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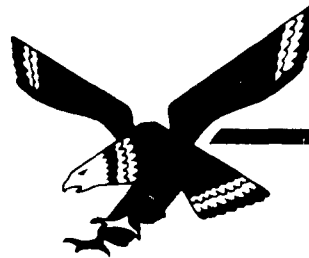# UNCLASSIFIED

MONTE CARLO AIR-SCATTERING DATA FOR MONOENERGETIC FAST NEUTRONS FROM POINT ISOTROPIC SOURCES

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MONTE CARLO AIR-SCATTERING DATA FOR MONOENERGETIC FAST NEUTRONS FROM POINT ISOTROPIC SOURCES

C. A. DIFFEY

ABSTRACT

The Monte Carlo fast-neutron air-scattering data
presented in FZK-9-147, Volumes I and II, have been inte-
grated to obtain the angular distributions and energy
spectra for a point isotropic source emitting one neutron
per second at a given energy E_o. These calculations were
performed for source-detector separations of 10, 35, 64,
and 100 feet and for initial neutron energies of 0.33,
1.1, 2.7, 4.0, 6.0, 8.0, 10.9, and 14.0 Mev.

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and 78.

LIST OF TABLES

I. INTRODUCTION

Air scattering is the principal process by which
neutrons leaving a source in air are transported to other
positions some distance away. In order to make comprehen-
sive shield-design studies, one must know the energy and
angular distributions of the scattered neutron flux at the
positions of interest. It is possible to design and apply
Monte Carlo procedures which solve this problem directly
for specific cases; however, it is often more practical to
use the Monte Carlo procedures to generate data in a suita-
ble parametric form so as to be able to apply the results
without having to run a new problem for each new source
term.

This report presents parametric air-scattering data
for isotropic sources of monoenergetic fast neutrons. For
source energies of 0.33, 1.1, 2.7, 4.0, 6.0, 8.0, 10.9,
and 14.0 Mev, the angular distributions of the scattered
flux and dose rate and the energy spectra of the scattered
flux are given for source-receiver separation distances of
10, 35, 64, and 100 feet.

The data presented here were computed from the results
of Wells' Monte Carlo parameter study of neutron scattering
for directional point sources in an infinite, homogeneous
medium of air (Refs. 1 and 2). A straight-forward integration

procedure was formulated and programmed in Fortran for the IBM-704 in order to carry out the calculations. The procedure was designed to utilize directly the punched card output of Wells' Monte Carlo calculations.

Section II of this report deals with the geometry of the neutron-scattering problem while Section III describes the Monte Carlo data used in these calculations. The integration scheme and Fortran procedure are discussed in Section IV and the results are presented in Section V in the form of tables and graphs giving the angular distributions of the scattered flux and dose rate and the energy spectra of the scattered flux.

II. NEUTRON SCATTERING GEOMETRY

The geometry (Fig. 1) consists of a neutron source S located in an infinite, homogeneous medium of air. A detector D of unit cross section is located a distance $\underline{a}$ from the source S. The neutron current leaving the source is described by a polar angle K and an azimuthal angle $\emptyset$. The polar angle is measured with respect to the source-detector axis while the azimuthal angle is the angle between the positive y axis and the projection of the neutron direction on the y,z plane. The azimuthal angle is measured in a clockwise direction.

The detector angle β is the polar angle between the direction of the incoming neutron flux and the source-detector axis. The azimuthal angle $\emptyset'$ at the detector is defined in the same manner as $\emptyset$. The neutron current of energy E_0 moving in the direction $(K,\emptyset)$ at a point Q on the surface of a unit sphere about S is defined as $S(K,\emptyset,E_0)$. The number of neutrons passing through a surface element dA at Q per unit time is given by $S(K,\emptyset,E_0)dA$.
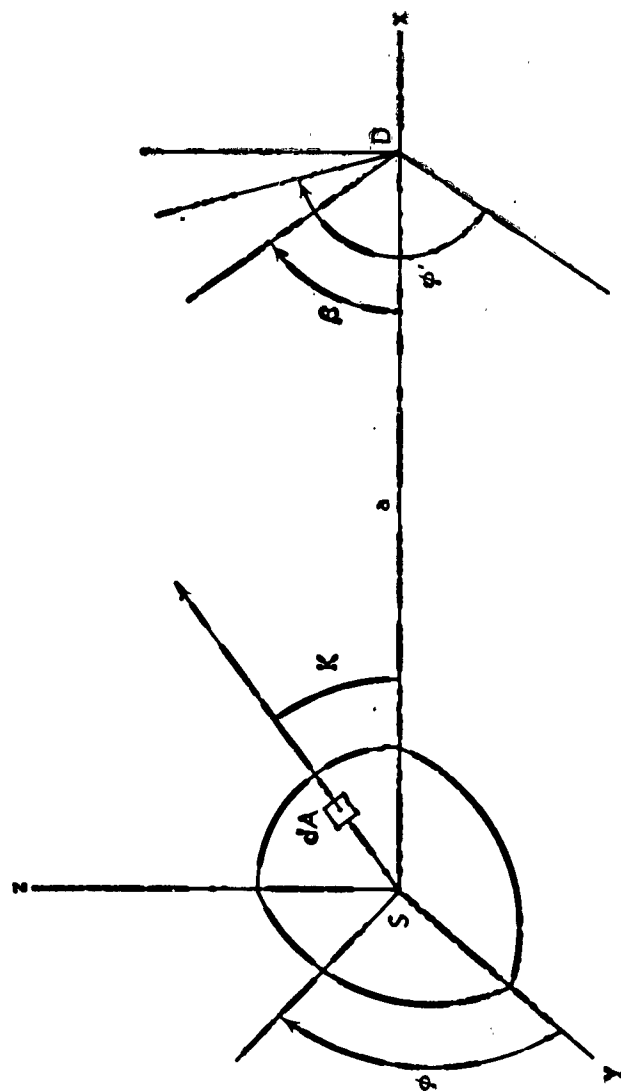
FIGURE 2. SCATTERING GEOMETRY

III. THE MONTE CARLO DATA

Since the generation of the Monte Carlo data used in these calculations has been thoroughly described by Wells in Reference 1, only a brief review is needed here.

The effects of elastic scattering, inelastic scattering and absorption were taken into account in the Monte Carlo air-scattering calculations. The neutron cross sections for nitrogen and oxygen used in these calculations were taken from data giving by Lustig, Goldstein, and Kalos (Refs. 3 and 4). The neutron cross sections for air were computed on the basis of a mixture of 78% nitrogen and 22% oxygen and a density of 5.37×10^{19} atoms/cc or 0.1293×10^{-2} gm/cc. The flux-to-dose conversion factors $F(E)$ used in computing the tissue dose rates were those calculated by Hurst and Ritchie (Ref. 5).

The calculated results presented in Reference 2 represent the angular distributions of the neutron flux, the angular distributions of the tissue dose rate, and the energy distribution of the neutron flux at the detector D for a point monodirectional source S emitting one neutron per second of energy E_o in the given direction $(K,\emptyset)$. The calculations were performed for four separation distances of 10, 35, 64, and 100 feet, eight initial neutron energies of 0.33, 1.1, 2.7, 4.0, 6.0, 8.0, 10.9 and 14.0 Mev, and

eight source angles, K, of 5, 15, 30, 60, 90, 120, 150, and
180 degrees. This represents a total of 64 different source
descriptions and four source-detector separations which may
be combined to describe any arbitrary source distribution.

The scattered flux reaching the detector is sorted
into eighteen equal angle intervals, β_i, and ten arbitrary
energy intervals, E_j. $N(K,\emptyset,E_o,a,\beta_i,E_j)$ is defined as the
fraction of neutrons with initial energy E_o and direction
$(K,\emptyset)$ passing through the incremental area dA per unit time
which reach the detector at a distance a in the detector
angle increment β_i with energy in the energy group E_j.
$D(K,\emptyset,E_o,a,\beta_i)$ is defined as the neutron dose rate due to
neutrons with initial energy E_o and direction $(K,\emptyset)$ which
arrive at the detector D a distance a from the source in
the detector angle increment β_i. Since there is symmetry
about the source-detector axis and both source and detector
are located in the infinite homogeneous medium, angular
distributions at the detector are independent of both source
and detector azimuthal angle. Therefore, the data are not
sorted with respect to this angle, and the variable $\emptyset$ is
dropped from the expressions.

The following quantities are tabulated in Reference 2:
$N(K,E_o,a,\beta_i,E_j)$, $N(K,E_o,a,E_j)$, $N(K,E_o,a,\beta_i)$, $D(,E_o,a,\beta_i)$,
$N(K,E_o,a)$, and $D(K,E_o,a)$. $N(K,E_o,a,\beta_i,E_j)$ is tabulated in
the form of an 18 x 10 matrix for each of the four values

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of a, eight values of E_O, and eight values of K. In addition to the 18 x 10 matrices, the energy distribution of the air-scattered flux is given by

$$N(K,E_O,a,E_j) = \sum_{i=1}^{18} N(K,E_O,a,\beta_i,E_j);$$ (1a)

the angular distributions of the air-scattered flux is given by

$$N(K,E_O,a,\beta_i) = \sum_{j=1}^{10} N(K,E_O,a,\beta_i,E_j).$$ (2a)

The dose rate $D(K,E_O,a,\beta_i)$ is calculated and stored as a function of K, E_O, a and β_i. It is printed as a function of angle β for each combination of K, E_O, and a.

The total scattered flux and dose rate are given, respectively, by

$$N(K,E_O,a) = \sum_{i=1}^{18} \sum_{j=1}^{10} N(K,E_O,a,\beta_i,Ej),$$ (4a)

and

$$D(K,E_O,a) = \sum_{i=1}^{18} D(K,E_O,a,\beta_i).$$ (5a)

The units of the scattered fluxes are neutrons/cm^2-sec per source neutron/sec and those of the scattered dose rates are rem/hr per source neutron/sec.

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The total scattered flux from a source $S(K,E_o)$ reaching the detector D at a distance a is

$$N(E_o,a) = \int_0^\pi \int_0^{2\pi} S(K,E_o)N(K,E_o,a) \sin K \, dK \, d\phi \tag{6a}$$

$$= 2\pi \int_0^\pi S(K,E_o)N(K,E_o,a) \sin K \, dK. \tag{6b}$$

Similarly, the total scattered dose rate is

$$D(E_o,a) = \int_0^\pi \int_0^{2\pi} S(K,E_o)D(K,E_o,a) \sin K \, dK \, d\phi \tag{7a}$$

$$= 2\pi \int_0^\pi S(K,E_o)D(K,E_o,a) \sin K \, dK. \tag{7b}$$

Similarly, the angular distributions of flux and dose rate are given, respectively, by

$$N(E_o,a,\beta) = \int_0^\pi \int_0^{2\pi} S(K,E_o)N(K,E_o,a,\beta) \sin K \, dK \, d\phi \tag{8a}$$

$$= 2\pi \int_0^\pi S(K,E_o)N(K,E_o,a,\beta) \sin K \, dK, \tag{8b}$$

and

$$D(E_o,a,\beta) = \int_0^\pi \int_0^{2\pi} S(K,E_o)D(K,E_o,a,\beta) \sin K \, dK \, d\phi \tag{9a}$$

$$= 2\pi \int_0^\pi S(K,E_o)D(K,E_o,a,\beta) \sin K \, dK. \tag{9b}$$

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The energy distribution of the scattered flux is given by

$$N(a,E_0E) = \int_0^\pi \int_0^{2\pi} S(K,E_0)N(K,E_0,a,E) \, \sin K \, dK \, d\phi \qquad (10a)$$

$$= 2\pi \int_0^\pi S(K,E_0)N(K,E_0,a,E) \, \sin K \, dK. \qquad (10b)$$

An analysis of the Monte Carlo data showed that the standard deviation of the total scattered flux $N(K,E_0,a)$ was less than $0.1 \, N(K,E_0,a)$ in 60.2% of the problems, less than $0.15 \, N(K,E_0,a)$ in 91.4% of the problems, and in only 2.73% of the problems did the standard deviation exceed $0.2 \, N(K,E_0,a)$.

IV. INTEGRATION OF THE MONTE CARLO DATA

Since the quantities $N(K,E_0,a,\beta)$, $D(K,E_0,a,\beta)$, and $N(K,E_0,a,E)$ are not given in terms of analytic functions, it is necessary to devise some integration scheme that will produce accurate results. For the purpose of numerical integrations, Equations 6b, 7b, 8b, 9b, and 10b become, in turn

$$N(E_0,a) = 2\pi \sum_{n=1}^{8} S(K_n,E_0)N(K_n,E_0,a) \sin K_n (\Delta K_n), \qquad (6c)$$

$$D(E_0,a) = 2\pi \sum_{n=1}^{8} S(K_n,E_0)D(K_n,E_0,a) \sin K_n (\Delta K_n), \qquad (7c)$$

$$N(E_0,a,\beta_i) = 2\pi \sum_{n=1}^{8} S(K_n,E_0)N(K_n,E_0,a,\beta_i) \sin K_n (\Delta K_n), \qquad (8c)$$

$$D(E_0,a,\beta_i) = 2\pi \sum_{n=1}^{8} S(K_n,E_0)D(K_n,E_0,a,\beta_i) \sin K_n (\Delta K_n), \qquad (9c)$$

$$N(E_0,a,E_j) = 2\pi \sum_{n=1}^{8} S(K_n,E_0)N(K_n,E_0,a,E_j) \sin K_n (\Delta K_n), \qquad (10c)$$

where $i=1, 2, \ldots, 18$ and $j = 1, 2, \ldots, 10$.

The calculations presented here were performed for a point isotropic source emitting 1 neutron/sec with energy E_0. Therefore,

$$S(K, E_0) = \text{constant} = \frac{1}{4\pi} \qquad (11)$$

The values of N and D were taken from the tables in Reference 2.

The summations in Equations 6c and 7c were performed using a machine code designed primarily for matrix calculations but adaptable for these calculations.

An IBM Fortran procedure was written to provide for the integration of Equations 8c, 9c, and 10c with a nonisotropic source term depending only on the polar angle K. The isotropic case presented here is a special case of this more general source term. A copy of the Fortran program is shown in the Appendix. Certain subroutines, such as SETUP, END 9, LIB 1, LIB, and END(2,1), are special General Dynamics/Fort Worth subroutines. However, the basic program should remain the same anywhere.

The values of (ΔK_n) used in the numerical integration were chosen after considering which histogram would best represent the smooth curve for integration purposes. The histogram-fit tends to overestimate in the first 20 degrees and underestimate in the last 20 degrees.

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The validity of the integration scheme was checked by comparing the results of numerical integration with those obtained using a planimeter on smooth curves drawn through the eight points. In the seven cases chosen at random for comparison, the numerical integration overestimated the result by less than 2.5%. In all cases, the numerical integration overestimated the result.

V. RESULTS

Equations 6c, 7c, 8c, 9c, and 10c have been numerically integrated for a point isotropic source. The results are presented in this section in both tabulated and graphical form.

The scattered neutron flux, $N(E_o,a,E_j)$, Tables II through V, and the neutron flux per Mev, $N(E_o,a,E_j)/\Delta E_j$, Tables VI through IX, have been tabulated as functions of E_o, the initial energy, and $(E_m)_j$. The subscript m indicates that $(E_m)_j$ is the lower limit of the jth energy interval. $(E_m)_j$ for each E_o is shown in Table I. The minimum energy cutoff for each source energy is given in Table I by $(E_m)_j$ for each source energy. ΔE_j is the width of the jth energy group. The scattered neutron flux, $N(E_o,a,E_j)$, is reported in units of

$$\frac{neutrons/cm^2\text{-}sec}{source\ neutron/sec}$$

and the neutron flux per Mev, $N(E_o,a,E_j)/\Delta E_j$ in units of

$$\frac{neutrons/cm^2\text{-}sec/Mev}{source\ neutron/sec} \ .$$

The angular distributions of the scattered neutron flux and dose rate in a 10° interval of β, $N(E_o,a,\beta_l)$ and $D(E_o,a,\beta_l)$, are tabulated as functions of E_o and β_l in Tables X through XIII and XIV through XVII, respectively,

where β_i is the upper limit of the ith angular interval. The units here are

$$\frac{\text{neutrons/cm}^2\text{-sec in } 10^o \text{ interval}}{\text{source neutron/sec}}$$

and

$$\frac{\text{rem/hr in } 10^o \text{ interval}}{\text{source neutron/sec}} \qquad .$$

The quantities $N(E_o,a,E_j)/\Delta E_j$, $N(E_o,a,\beta_i)$, $D(E_o,a,\beta_i)$, $N(E_o,a)$, and $D(E_o,a)$ are shown in Figures 2-9, 10-17, 18-25, 26 and 27, respectively.

The angular distributions are plotted as function of $\widetilde{\beta}_i$, the midpoint of the angular interval β_{i-1} to β_i.

In Figures 2 through 9, the energy spectrum is plotted against the midpoint of the energy group E_j.

The total scattered flux and dose rate, $N(E_o,a)$ and $D(E_o,a)$, respectively, are shown in Figures 26 and 27 as functions of E_o for each of the four separation distances considered.

The scattered dose rates for source energies of 1.1 Mev or greater result from neutrons with energies greater than the minimum energy used for each source energy, but the scattered dose rates for the 0.33 Mev source are those resulting from neutrons with energies greater than 0.22 Mev.

TABLE I. LOWER BOUNDS OF THE ENERGY GROUPS USED TO DEFINE THE
SCATTERED NEUTRON FLUX FOR EACH SOURCE ENERGY

$$(E_m)_j \ (\text{Mev})$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.070	0.20	0.60	0.33	0.33	0.33	0.33	0.33
2	0.096	0.29	0.90	0.50	0.75	1.75	1.75	1.75
3	0.122	0.38	1.10	0.80	1.50	2.50	2.50	2.50
4	0.148	0.47	1.30	1.20	2.25	3.00	3.75	3.50
5	0.174	0.56	1.50	1.60	2.75	3.50	4.75	4.50
6	0.200	0.65	1.70	2.00	3.50	4.50	5.50	5.50
7	0.226	0.74	1.90	2.40	4.00	5.50	6.75	6.75
8	0.252	0.83	2.10	2.80	4.50	6.00	8.00	8.00
9	0.278	0.92	2.30	3.20	5.00	6.50	9.00	10.0
10	0.304	1.01	2.50	3.60	5.50	7.25	10.0	12.0

TABLE II. TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 10 Feet

$$[(\text{neutrons}/\text{cm}^2\text{-sec})/(\text{source neutron/sec})]$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.1122-12	0.	0.5394-12	0.1167-13	0.1128-08	0.2512-08	0.4350-08	0.8118-09
2	0.2790-10	0.4669+10	0.2064-10	0.5279-12	0.1878-08	0.2370-08	0.2482-09	0.3264-08
3	0.2937-09	0.6109-10	0.1182-09	0.7102-10	0.1409-08	0.1955-08	0.2258-08	0.1626-08
4	0.7100-09	0.2407-09	0.1914-09	0.3504-10	0.1764-09	0.1092-08	0.4100-08	0.1231-09
5	0.3901-08	0.7973-09	0.6295-09	0.8408-10	0.2141-09	0.6667-09	0.2416-08	0.5218-09
6	0.2775-08	0.2744-09	0.2014-08	0.6066-09	0.4960-09	0.4670-09	0.2055-08	0.3969-08
7	0.1578-07	0.5649-08	0.8703-08	0.2911-08	0.3923-09	0.3684-09	0.5132-10	0.3698-08
8	0.2731-07	0.5376-07	0.1313-07	0.2503-07	0.1420-07	0.6221-08	0.4220-09	0.2806-08
9	0.2144-07	0.1407-07	0.8017-08	0.1431-07	0.6063-08	0.4763-08	0.2298-08	0.5662-09
10	0.2060-07	0.1475-07	0.8423-08	0.1322-07	0.1249-07	0.1104-07	0.8306-08	0.1423-07

$0.1122\text{-}12 = 1.122 \times 10^{-13}$

TABLE III. TOTAL SCATTERED NEUTRON FLUX
Separation Distance - 35 Feet

$$[(\text{neutrons}/\text{cm}^2\text{-sec})/(\text{source neutron/sec})]$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.7994-13	0.	0.1401-11	0.7245-14	0.5410-09	0.1047-08	0.1638-08	0.4155-09
2	0.2608-10	0.7550-12	0.2016-10	0.496-1?	0.6236-09	0.9556-09	0.8950-10	0.8967-09
3	0.1916-09	0.3998-10	0.1217-09	0.5142-10	0.1875-09	0.6682-09	0.7220-09	0.3803-09
4	0.4971-09	0.2309-09	0.1843-09	0.1496-10	0.9951-10	0.2960-09	0.8941-09	0.1023-09
5	0.124?-08	0.7289-09	0.3327-09	0.8966-10	0.1712-09	0.3182-09	0.6678-09	0.2930-09
6	0.2157-08	0.1620-08	0.5705-09	0.5021-09	0.2181-09	0.1877-09	0.4181-09	0.1457-08
7	0.3934-08	0.2279-08	0.2448-08	0.1370-08	0.4158-09	0.1005-09	0.1688-09	0.9020-09
8	0.6643-08	0.1272-07	0.3433-08	0.6336-08	0.3388-08	0.1595-08	0.9787-09	0.6838-09
9	0.4941-08	0.3213-08	0.2513-08	0.4133-08	0.1582-08	0.1165-08	0.8691-09	0.1459-08
10	0.5449-08	0.3962-08	0.2078-08	0.3972-08	0.3485-08	0.3118-08	0.2571-08	0.4062-08

0.7994-13 = 7.994x10^-14

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TABLE IV. TOTAL SCATTERED NEUTRON FLUX
Separation Distance = 64 Feet

$$\left[\frac{(\text{neutrons/cm}^2\text{-sec})}{(\text{source-neutron/sec})}\right]$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.6086-13	0.	0.1209-11	0.1068-14	0.3648-09	0.6106-09	0.6175-09	0.2784-09
2	0.2306-10	0.7801-12	0.2046-10	0.5026-12	0.5169-09	0.4549-09	0.6783-10	0.4817-09
3	0.1588-09	0.3263-10	0.3247-09	0.3786-10	0.1105-09	0.3527-09	0.4518-09	0.2851-09
4	0.4821-09	0.2229-09	0.1417-09	0.1988-10	0.9799-10	0.2062-09	0.4905-09	0.7119-10
5	0.7371-09	0.5434-09	0.2486-09	0.1052-09	0.1588-09	0.2452-09	0.3351-09	0.1776-09
6	0.1148-08	0.1176-08	0.4671-09	0.3451-09	0.1897-09	0.2141-09	0.2462-09	0.6094-09
7	0.2176-08	0.1685-08	0.1179-08	0.1051-08	0.3678-09	0.1174-09	0.9096-10	0.6088-09
8	0.3306-08	0.5393-08	0.1683-08	0.3358-08	0.1700-08	0.8854-09	0.4640-09	0.3930-09
9	0.2528-08	0.1696-08	0.1333-08	0.2046-08	0.1054-08	0.7237-09	0.4625-09	0.7167-09
10	0.2497-08	0.1919-08	0.1162-08	0.1929-08	0.1599-08	0.1570-08	0.1312-08	0.1845-08

0.6086-13 = 6.086x10^{-14}

TABLE V.   TOTAL SCATTERED NEUTRON FLUX
Separation Distance – 100 Feet

$$\left[(\text{neutrons}/\text{cm}^2\text{-sec})/(\text{source neutron}/\text{sec})\right]$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.4276-13	0.	0.9111-11	0.1103-14	0.1755-09	0.4964-09	0.5456-09	0.1845-09
2	0.2850-10	0.4220-12	0.2009-10	0.2177-11	0.2877-09	0.3141-09	0.5019-10	0.3015-09
3	0.1530-09	0.4332-10	0.8127-10	0.1287-10	0.6905-10	0.2214-09	0.4010-09	0.1235-09
4	0.3945-09	0.2484-09	0.1386-09	0.2256-10	0.1065-09	0.1082-09	0.2241-09	0.6058-10
5	0.7213-09	0.4144-09	0.2417-09	0.8228-10	0.1350-09	0.0597-10	0.2154-09	0.1642-09
6	0.6783-09	0.8503-09	0.3731-09	0.7410-09	0.1602-09	0.1150-09	0.1281-09	0.4271-09
7	0.1348-08	0.9637-09	0.7032-09	0.6224-09	0.2867-09	0.7642-10	0.5522-10	0.4002-09
8	0.1754-08	0.2784-08	0.9533-09	0.1871-08	0.5475-09	0.4564-09	0.2474-09	0.2154-09
9	0.1337-08	0.1175-08	0.7217-09	0.1149-08	0.6040-09	0.4504-09	0.6146-09	0.4028-09
10	0.1343-08	0.1165-08	0.6462-08	0.1170-08	0.656-08	0.5477-08	0.7774-09	0.1217-08

0.4276-13 = 0.276x10⁻¹⁴

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TABLE VI. TOTAL SCATTERED NEUTRON FLUX PER MEV
Separation Distance - 10 Feet

$$\left[\frac{(\text{neutron}/\text{cm}^2\text{-sec-Mev})}{(\text{source neutron}/\text{sec})}\right]$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.4315-11	0.0000-00	0.1798-11	0.6865-13	0.2686-08	0.1769-08	0.3063-08	0.5717-09
2	0.1073-08	0.8372-11	0.1032-09	0.1760-11	0.2493-08	0.3160-08	0.3309-09	0.4352-08
3	0.1145-07	0.6788-09	0.5910-09	0.1776-09	0.1879-08	0.3910-08	0.1886-08	0.1626-08
4	0.2731-07	0.2674-08	0.9570-09	0.8760-10	0.3526-09	0.2184-08	0.4099-08	0.1231-09
5	0.1500-06	0.8859-08	0.3148-08	0.2102-09	0.2855-09	0.6667-09	0.3221-08	0.5218-09
6	0.1066-06	0.3048-07	0.1007-07	0.1517-08	0.9920-09	0.4669-09	0.2444-08	0.3175-08
7	0.6069-06	0.6277-07	0.4352-07	0.7278-08	0.7846-09	0.7368-09	0.4106-10	0.2958-08
8	0.1050-05	0.5967-06	0.6565-07	0.6258-07	0.2840-07	0.1245-07	0.4220-08	0.1403-08
9	0.8246-06	0.1563-06	0.4009-07	0.3578-07	0.1213-07	0.6351-08	0.2298-08	0.2831-08
10	0.7923-06	0.1639-06	0.4212-07	0.3305-07	0.2498-07	0.1471-07	0.9229-08	0.7115-08

0.4315-11 = 4.315x10⁻¹²

TABLE VII. TOTAL SCATTERED NEUTRON FLUX PER MEV

Separation Distance - 35 Feet

$$\left[\left(\text{neutron/cm}^2\text{-sec-Mev}\right)/\left(\text{source neutron/sec}\right)\right]$$

Energy Group Index J	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.3075-12	0.0000-00	0.4670-11	0.4261-14	0.1288-08	0.7373-09	0.1153-08	0.2925-09
2	0.9262-09	0.8388-11	0.1008-09	0.1656-11	0.8314-09	0.1274-08	0.1327-09	0.1196-08
3	0.7369-08	0.4442-09	0.6235-09	0.1286-09	0.2500-09	0.1336-08	0.5776-09	0.3803-09
4	0.1912-07	0.2566-08	0.9215-09	0.3740-10	0.1990-09	0.5920-09	0.8941-09	0.1023-09
5	0.4796-07	0.8209-08	0.1914-08	0.2241-09	0.2283-09	0.3182-09	0.8904-09	0.2930-09
6	0.8296-07	0.1800-07	0.2853-08	0.1255-08	0.4362-09	0.1877-09	0.3345-09	0.1166-08
7	0.1513-06	0.2532-07	0.1224-07	0.3425-08	0.8316-09	0.2010-09	0.1350-09	0.7216-09
8	0.2555-06	0.1413-06	0.1717-07	0.1584-07	0.6776-08	0.3190-08	0.9789-09	0.3419-09
9	0.1900-06	0.3570-07	0.1257-07	0.1033-07	0.3164-08	0.1553-08	0.8690-09	0.7295-09
10	0.2096-06	0.4402-07	0.1039-07	0.9930-08	0.6970-08	0.4157-08	0.2856-08	0.2031-08

$0.3075-12 = 3.075 \times 10^{-13}$

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TABLE VIII. TOTAL SCATTERED NEUTRON FLUX PER MEV

Separation Distance = 64 Feet

$$\left[\frac{(\text{neutron/cm}^2\text{-sec-Mev})}{(\text{source neutron/sec})}\right]$$

Energy Group Index J	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.2341-11	0.0000-00	0.4030-11	0.6282-14	0.8686-09	0.4300-09	0.5757-09	0.1960-09
2	0.8869-09	0.8667-11	0.1023-09	0.1675-11	0.6892-09	0.6065-09	0.9042-10	0.6423-09
3	0.6104-08	0.3626-09	0.1624-08	0.9465-10	0.1473-09	0.7052-09	0.3854-09	0.2851-09
4	0.1854-07	0.2477-08	0.7085-09	0.4970-10	0.1960-09	0.4124-09	0.4905-09	0.7119-10
5	0.2835-07	0.6037-08	0.1243-08	0.2628-09	0.2117-09	0.2452-09	0.4468-09	0.1776-09
6	0.4415-07	0.1307-07	0.2336-08	0.8628-08	0.3790-09	0.2141-09	0.1970-09	0.4875-09
7	0.8369-07	0.1883-07	0.5895-08	0.2628-08	0.7354-09	0.2348-09	0.7277-10	0.4870-09
8	0.1269-06	0.5992-07	0.8415-08	0.8395-08	0.3400-08	0.1771-08	0.4840-09	0.1965-09
9	0.9723-07	0.1884-07	0.6610-08	0.5115-08	0.1108-08	0.9649-09	0.4625-09	0.3583-09
10	0.9604-07	0.2132-07	0.5810-08	0.4823-08	0.3198-08	0.2093-08	0.1458-08	0.9225-09

0.2341-11 = 2.341×10^{-12}

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TABLE IX. TOTAL SCATTERED NEUTRON FLUX PER MEV

Separation Distance = 100 Feet

$$\left[\frac{\text{neutron}/\text{cm}^2\text{-sec-Mev}}{\text{source neutron}/\text{sec}}\right]$$

Energy Group Index j	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.1644-11	0.0000-00	0.3703-11	0.6488-14	0.4244-09	0.3495-09	0.3842-09	0.1299-09
2	0.1096-08	0.4689-11	0.1005-09	0.7257-11	0.3828-09	0.4188-09	0.7892-10	0.4020-09
3	0.5885-08	0.4813-09	0.4063-09	0.3218-10	0.9205-10	0.4428-09	0.3208-09	0.1335-09
4	0.1517-07	0.2760-08	0.6930-09	0.5640-10	0.2130-09	0.2164-09	0.2241-09	0.6058-10
5	0.2774-07	0.4604-08	0.1209-08	0.2057-09	0.1800-09	0.8987-10	0.2872-09	0.1242-09
6	0.2609-07	0.9447-08	0.1866-08	0.8675-09	0.3204-09	0.1180-09	0.1025-09	0.3417-09
7	0.5181-07	0.1071-07	0.3516-08	0.2056-08	0.5774-09	0.1576-09	0.4418-10	0.3202-09
8	0.6746-07	0.3093-07	0.4767-08	0.4678-08	0.1895-08	0.9728-09	0.2474-09	0.1097-09
9	0.5142-07	0.1306-07	0.3609-08	0.2873-08	0.1208-08	0.6004-09	0.3146-09	0.2049-09
10	0.5165-07	0.1294-07	0.3201-08	0.2923-08	0.199-08	0.1264-08	0.8638-09	0.6085-09

0.1644-11 = 1.644×10⁻¹²

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TABLE X . ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX

Separation Distance - 10 Feet

$$[(\text{neutrons}/\text{cm}^2\text{-sec})/(\text{source neutron}/\text{sec})]$$

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.1122-07	0.9942-08	0.4957-08	0.6558-08	0.6104-08	0.5060-08	0.4456-08	
20	0.1054-07	0.1060-07	0.4611-08	0.6806-08	0.5372-08	0.5074-08	0.3246-08	
30	0.9124-08	0.6955-08	0.5428-08	0.5212-08	0.4770-08	0.4505-08	0.2489-08	0.3782-08
40	0.1019-07	0.6924-08	0.4250-08	0.4599-08	0.4570-08	0.2746-08	0.2655-08	0.2334-08
50	0.5104-08	0.7208-08	0.3178-08	0.4972-08	0.3241-08	0.2651-08	0.1565-08	0.1544-08
60	0.6357-08	0.6285-08	0.3880-08	0.4228-08	0.2061-08	0.1574-08	0.1246-08	0.1411-08
70	0.6508-08	0.5420-08	0.2317-08	0.3174-08	0.2347-08	0.1426-08	0.1516-08	0.2283-08
80	0.5911-08	0.6200-08	0.1905-08	0.3500-08	0.2761-08	0.1467-08	0.1101-08	0.1501-08
90	0.5794-08	0.4625-08	0.2566-08	0.2769-08	0.2373-08	0.1186-08	0.2144-08	0.1035-08
100	0.4296-08	0.4814-08	0.1853-08	0.7932-08	0.1362-08	0.1015-08	0.1695-08	0.1848-08
110	0.4090-08	0.5075-08	0.1605-08	0.7246-08	0.1045-08	0.1655-08	0.1340-08	0.7659-09
120	0.3297-08	0.4353-08	0.1184-08	0.2780-08	0.1701-08	0.7857-09	0.1032-08	0.8606-09
130	0.1838-08	0.2451-08	0.1081-08	0.1211-08	0.4333-09	0.6927-09	0.5874-09	0.5031-09
140	0.1719-08	0.3227-08	0.6343-09	0.1643-09	0.7120-09	0.5468-09	0.4502-09	0.5939-09
150	0.1596-08	0.2491-08	0.8437-09	0.1336-08	0.5654-09	0.5291-09	0.5254-09	0.5987-09
160	0.1375-08	0.2004-08	0.4178-09	0.1581-08	0.3716-09	0.3282-09	0.2771-09	0.5334-09
170	0.6936-09	0.1046-08	0.2567-09	0.5276-09	0.2150-09	0.1630-09	0.1938-09	0.2383-09
180	0.1763-09	0.4016-09	0.8089-10	0.1547-09	0.6936-10	0.5092-10	0.6550-10	0.6101-10

0.1122-07 = 1.122×10^{-8}

TABLE XI . ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX
Separation Distance = 35 Feet

[(neutrons/cm²-sec)/(source neutron/sec)]

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
	0.2985-08	0.1676-08	0.1389-08	0.2117-08	0.1773-08	0.1717-08	0.1635-08	0.2094-08
	0.2951-08	0.2360-08	0.1568-08	0.1199-08	0.1516-08	0.1244-08	0.1454-08	0.1625-08
	0.2792-08	0.2277-08	0.1255-08	0.1575-08	0.163-08	0.1181-08	0.1011-08	0.1104-08
	0.2375-08	0.366-08	0.1069-08	0.1356-08	0.1102-08	0.7625-09	0.7716-09	0.7057-09
	0.2365-09	0.1944-08	0.1027-03	0.1530-08	0.8159-09	0.7796-09	0.7732-09	0.7480-09
	0.1647-08	0.1569-08	0.9763-09	0.1119-08	0.8159-09	0.4075-09	0.5154-05	0.6119-09
	0.1613-08	0.1590-08	0.4571-09	0.4706-08	0.7162-08	0.4730-09	0.6654-05	0.9164-09
	0.1527-08	0.1447-08	0.6552-05	0.1720-09	0.4675-09	0.4717-09	0.4707-09	0.5055-09
	0.161-08	0.1505-08	0.7226-05	0.634-09	0.5651-09	0.3683-09	0.5716-05	0.4188-09
	0.1162-08	0.1808-08	0.5141-09	0.1562-09	0.4286-09	0.3684-09	0.4760-09	0.4541-09
	0.1086-08	0.1265-08	0.4465-09	0.177-09	0.4450-09	0.4565-09	0.4454-05	0.5305-05
	0.1507-08	0.5038-09	0.3860-09	0.643-09	0.2619-09	0.2516-09	0.2158-09	0.2644-
	0.451-09	0.2110-09	0.2566-09	0.5073-09	0.2774-09	0.9363-09	0.1749-09	
	0.7416-09	0.1913-09	0.2005-09	0.4412-09	0.1262-09	0.2487-09	0.1463-	
	0.4654-05	0.1529-09	0.1617-09	0.1452-09	0.1762-9	0.1271-09		
	0.1773-08	0.1616-08	0.8724-10	0.1512-0	0.1015-09	0.7342-10	0.4404-10	0.5673-10
	0.235-10	0.6167-10	0.3510-10	0.1573-	0.2411-10	0.1452-10	0.1960-10	0.7135-1

0.2985-08 = 2.985x10⁻⁹

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$$[(\text{neutrons}/\text{cm}^2\text{-sec})/(\text{source neutron}/\text{sec})]$$

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
1	0.1445-08	0.1291-08	0.7224-09	0.9822-09	0.9024-09	0.8263-09	0.9245-09	0.1038-08
2	0.1471-08	0.1258-08	0.7040-09	0.9266-09	0.8263-09	0.7580-09	0.7338-09	0.8059-09
3	0.1354-08	0.1205-08	0.6535-09	0.8988-09	0.6084-09	0.5679-09	0.5170-09	0.6273-09
4	0.1227-08	0.1053-08	0.7814-09	0.7660-09	0.4580-09	0.4663-09	0.4173-09	0.4947-09
5	0.1197-08	0.1483-08	0.5389-09	0.6667-09	0.4799-09	0.3683-09	0.2924-09	0.3868-09
6	0.9442-09	0.8502-09	0.4657-09	0.6445-09	0.3432-09	0.3996-09	0.2795-09	0.3307-09
7	0.8803-09	0.7493-09	0.4568-09	0.6775-09	0.3365-09	0.4103-09	0.2697-09	0.3153-09
8	0.7880-09	0.7260-09	0.3556-09	0.5836-09	0.3130-09	0.2565-09	0.2316-09	0.2662-09
9	0.7578-09	0.7211-09	0.3576-09	0.4172-09	0.2456-09	0.2395-09	0.2201-09	0.2362-09
10	0.8749-09	0.7095-09	0.3246-09	0.4547-09	0.7394-09	0.1967-09	0.1941-09	0.2209-09
11	0.4348-09	0.5938-09	0.2234-09	0.3945-09	0.2817-09	0.2641-09	0.2691-09	0.2036-09
12	0.3927-09	0.5000-09	0.1668-09	0.3110-09	0.1395-09	0.9276-09	0.1103-09	0.1441-09
13	0.4509-09	0.4402-09	0.1792-09	0.4457-09	0.2125-09	0.1134-09	0.1249-09	0.1029-09
14	0.2934-09	0.3820-09	0.1375-09	0.2467-09	0.1108-09	0.1322-09	0.8959-10	0.1328-09
15	0.2774-09	0.3095-09	0.1218-09	0.2373-09	0.7232-10	0.8370-10	0.7040-10	0.7653-10
16	0.1457-09	0.2258-09	0.8080-10	0.1199-09	0.6627-10	0.1272-09	0.6981-10	0.4807-10
17	0.8827-10	0.1403-09	0.2718-10	0.9904-10	0.9400-11	0.3386-10	0.2708-10	0.2823-10
18	0.2796-10	0.2962-10	0.9061-11	0.2091-10		0.1831-10	0.7067-11	0.9389-11

0.1445-08 = 1.445x10⁻⁹

TABLE XIII. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON FLUX
Separation Distance – 100 Feet

$$[(neutrons/cm^2\text{-}sec)/(source\ neutron/sec)]$$

	Source Energy (Mev)							
Detector Angular Interval (degrees)	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
	0.8001-09	0.7332-09	0.4817-09	0.5678-09	0.6540-09		0.?103-09	0.6298-09
	0.7540-09	0.7368-09	0.4401-09	0.5687-09	0.4634-09		0.?125-09	0.5366-09
		0.6932-09	0.3929-09	0.5655-09	0.4337-09			0.3771-09
	0.5172-09	0.6226-09	0.3519-09	0.5679-09	0.4337-09		0.468?-09	0.3231-09
	0.6620-09	0.5291-09	0.3279-09	0.4656-09	0.7525-09	0.2145-09	0.?192-09	0.2969-09
		0.5719-09	0.2900-09	0.4267-09	0.2712-09		0.?598-09	0.1892-09
	0.4812-09	0.5053-09	0.2303-09	0.3089-09	0.2382-09	0.2132-09	0.?592-09	0.2140-09
	0.4513-09	0.6461-09	0.2714-09	0.3104-09	0.2321-09	0.1745-09	0.?783-09	0.1644-09
		0.5617-09	0.1821-09	0.3280-09	0.1672-09	0.1418-09	0.?872-09	0.1456-09
		0.3602-09	0.1781-09	0.2713-09	0.1899-09	0.1404-09	0.?152-09	0.1556-09
	0.7130-09	0.3872-09	0.1604-09	0.2530-09	0.2723-09	0.9645-10	0.?116-09	0.1325-09
		0.2593-09	0.1174-09	0.2039-09	0.1558-09	0.1100-09	0.4729-10	0.7965-10
	0.2427-09	0.7487-09	0.1680-09	0.1733-09	0.1255-09	0.7285-10	0.7919-10	0.6276-10
	0.1370-09	0.2123-09	0.7924-10	0.1819-09	0.1267-09	0.7065-10	0.6165-10	0.6496-10
	0.1366-09	0.1586-09	0.1072-09	0.1391-09	0.8017-10	0.1030-09	0.3785-10	0.4268-10
	0.1935-09	0.1358-09	0.7867-10	0.8375-10	0.6185-10	0.4762-10	0.4451-10	0.3769-10
	0.5668-10	0.1035-10	0.2268-10	0.5032-10	0.4968-10	0.7042-10	0.7647-10	0.2034-10
	0.5696-11	0.1351-10	0.4655-11	0.8472-11	0.4658-11	0.4584-11	0.4059-11	0.4775-11

0.8001-09 = 8.001x10^{-10}

TABLE XIV. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE

Separation Distance – 10 Feet

$$[(rem/hr)/(source-neutron/sec)]$$

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.5081-12	0.8819-12	0.6244-12	0.1066-11	0.1107-11	0.1208-11	0.1037-11	0.1895-11
20	0.4753-12	0.9263-12	0.5761-12	0.1107-11	0.9700-12	0.1009-11	0.1158-11	0.1397-11
30	0.4011-12	0.7667-12	0.6691-12	0.8460-12	0.8641-12	0.6956-12	0.7675-12	0.9061-12
40	0.4051-12	0.5843-12	0.5221-12	0.7422-12	0.6423-12	0.5414-12	0.6126-12	0.7987-12
50	0.3057-12	0.5947-12	0.3803-12	0.7978-12	0.5852-12	0.5229-12	0.4304-12	0.6745-12
60	0.2623-12	0.5226-12	0.4769-12	0.6847-12	0.3725-12	0.7071-12	0.4812-12	0.5572-12
70	0.2681-12	0.4548-12	0.2768-12	0.5105-12	0.4167-12	0.2772-12	0.3115-12	0.5404-12
80	0.2301-12	0.5128-12	0.2261-12	0.5662-12	0.3934-12	0.2882-12	0.2323-12	0.3458-12
90	0.2433-12	0.3680-12	0.3172-12	0.4471-12	0.4022-12	0.2329-12	0.4491-12	0.2407-12
100	0.1693-12	0.3945-12	0.2182-12	0.4727-12	0.2447-12	0.1987-12	0.3573-12	0.4236-12
110	0.1632-12	0.4076-12	0.2212-12	0.3632-12	0.1866-12	0.3241-12	0.2883-12	0.1763-12
120	0.1296-12	0.3516-12	0.1386-12	0.4480-12	0.2994-12	0.1513-12	0.2211-12	0.2022-12
130	0.7431-13	0.1983-12	0.1317-12	0.2082-12	0.7225-13	0.1312-12	0.1202-12	0.1130-12
140	0.6860-13	0.2615-12	0.7394-13	0.2638-12	0.1281-12	0.1009-12	0.5248-13	0.1325-12
150	0.6438-13	0.1999-12	0.1038-12	0.2137-12	0.1012-12	0.1022-12	0.1101-12	0.1396-12
160	0.4147-13	0.1734-12	0.5185-13	0.2494-12	0.6592-13	0.6272-13	0.5757-13	0.7834-13
170	0.3001-13	0.9408-13	0.3304-13	0.8504-13	0.3755-13	0.3052-13	0.4160-13	0.5405-13
180	0.8234-14	0.3679-13	0.1064-13	0.2444-13	0.1219-13	0.9518-14	0.1426-13	0.1435-13

0.5081-12 = 5.081x10^-13

40

TABLE XV   ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE
Separation Distance - 35 Feet

$$[(rem/hr)/(source-neutron/sec)]$$

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.1321-12	0.2368-12	0.1742-12	0.3433-12	0.3207-12	0.3416-12	0.3611-12	0.5010-12
20	0.1267-12	0.2040-12	0.1690-12	0.2909-12	0.2732-12	0.2448-12	0.3112-12	0.3865-12
30	0.1183-12	0.1958-12	0.1528-12	0.2536-12	0.1960-12	0.2318-12	0.2171-12	0.2613-12
40	0.9623-13	0.1978-12	0.1306-12	0.2200-12	0.1985-12	0.1434-12	0.1547-12	0.1771-12
50	0.8775-13	0.1603-12	0.1283-12	0.2134-12	0.1411-12	0.1191-12	0.1623-12	0.1727-12
60	0.7571-13	0.1235-12	0.1162-12	0.1791-12	0.1091-12	0.1131-12	0.1085-12	0.1724-12
70	0.6514-13	0.1266-12	0.8287-13	0.1159-12	0.1073-12	0.8911-13	0.5689-13	0.2043-12
80	0.4986-13	0.1166-12	0.1115-12	0.1540-12	0.8634-13	0.8579-13	0.8394-13	0.1135-12
90	0.5749-13	0.1113-12	0.8685-13	0.1378-12	0.1464-13	0.7027-13	0.7580-13	0.9541-13
100	0.4175-13	0.1113-12	0.1271-12	0.1706-12	0.7381-13	0.7431-13	0.7308-13	0.1039-12
110	0.3828-13	0.1172-12	0.5286-13	0.1119-12	0.7852-13	0.6469-13	0.6822-13	0.6996-13
120	0.7427-13	0.7837-13	0.4260-13	0.6975-13	0.4535-13	0.4682-13	0.6428-13	0.6531-13
130	0.1657-13	0.7101-13	0.3418-13	0.7879-13	0.4846-13	0.4258-13	0.4415-13	0.5542-13
140	0.2390-13	0.7415-13	0.2626-13	0.6989-13	0.4268-13	0.3876-13	0.3616-13	0.5344-13
150	0.1720-13	0.5562-13	0.2425-13	0.5461-13	0.2757-13	0.3791-13	0.2729-13	0.2734-13
160	0.1028-13	0.3868-13	0.1726-13	0.4381-13	0.2728-13	0.2055-13	0.1460-13	0.2129-13
170	0.7314-14	0.3156-13	0.1096-13	0.2364-13	0.1722-13	0.1180-13	0.9576-14	0.1283-13
180	0.1963-14	0.5607-14	0.3570-14	0.6086-14	0.4169-14	0.4597-14	0.3847-14	0.3130-14

0.1321-12 = 1.321×10^{-13}

41

TABLE XVI. ANGULAR DISTRIBUTION OF TOTAL SCATTERED NEUTRON DOSE RATE
Separation Distance – 64 Feet

$$[(rem/hr)/(source\text{-}neutron/sec)]$$

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.6339-13	0.1139-12	0.9070-13	0.1590-12	0.1629-12	0.1740-12	0.2025-12	0.2480-12
20	0.6190-13	0.1083-12	0.8733-13	0.1492-12	0.1475-12	0.1477-12	0.1575-12	0.1900-12
30	0.5506-13	0.1018-12	0.8001-13	0.1431-12	0.1071-12	0.1052-12	0.1100-12	0.1408-12
40	0.4814-13	0.8657-13	0.9623-13	0.1220-12	0.8015-13	0.8824-13	0.8603-13	0.1133-12
50	0.4551-13	0.8192-12	0.6508-13	0.1056-12	0.8080-13	0.7050-13	0.5934-13	0.8586-13
60	0.3612-13	0.6865-13	0.5571-13	0.9023-12	0.9549-13	0.7431-13	0.5586-13	0.7199-13
70	0.3144-13	0.5943-13	0.5426-13	0.9056-12	0.5849-13	0.7239-13	0.5247-13	0.7045-13
80	0.2896-13	0.5760-13	0.4274-13	0.9138-13	0.5095-13	0.4725-13	0.4500-13	0.5764-13
90	0.2435-13	0.5700-13	0.4200-13	0.6572-13	0.4857-13	0.4285-13	0.4284-13	0.5196-13
100	0.2028-13	0.4630-13	0.3882-13	0.7146-13	0.4162-13	0.3656-13	0.3792-13	0.4689-13
110	0.1333-13	0.3853-13	0.2631-13	0.6196-13	0.5716-13	0.3711-13	0.2853-13	0.4379-13
120	0.1089-13	0.3461-13	0.1987-13	0.4844-13	0.3566-13	0.2194-13	0.2122-13	0.3022-13
130	0.0534-13	0.3010-13	0.2119-13	0.7027-13	0.2378-13	0.2027-13	0.2317-13	0.2179-13
140	0.1062-13	0.2470-13	0.1629-13	0.3889-13	0.3678-13	0.2371-13	0.1672-13	0.2852-13
150	0.7633-14	0.1961-13	0.1468-13	0.3695-13	0.1809-13	0.1554-13	0.1574-13	0.1625-13
160	0.5139-14	0.1184-12	0.9737-14	0.4894-13	0.1252-13	0.2369-13	0.1348-13	0.1045-13
170	0.3144-14	0.1184-12	0.2896-13	0.1466-13	0.1046-13	0.6411-14	0.4937-14	0.6132-14
180	0.8539-15	0.0549-14	0.2140-14	0.7317-14	0.4228-14	0.7362-14	0.1374-14	0.2003-14

0.6339-13 = 6.339x10⁻¹⁴

$$\left[\text{(rem/hr)}\middle/\text{(source-neutron/sec)}\right]$$

Detector Angular Interval (degrees)	Source Energy (Mev)							
	0.33	1.1	2.7	4.0	6.0	8.0	10.9	14.0
10	0.3489-13	0.6425-13	0.5977-13	0.5183-13	0.3954-13	0.1065-12	0.1133-12	0.1492-12
20	0.3275-13	0.6277-13	0.5444-13	0.5085-13	0.8561-13	0.8671-13	0.8934-13	0.1236-12
30	0.3386-13	0.5703-13	0.4814-13	0.5007-13	0.7470-13	0.7770-13	0.7000-13	0.8754-13
40	0.3148-13	0.6118-13	0.4274-13	0.8975-13	0.6046-13	0.5098-13	0.4867-13	0.7169-13
50	0.2214-13	0.4264-13	0.3925-13	0.7317-13	0.4703-13	0.4938-13	0.5355-13	0.6380-13
60	0.2180-13	0.4537-13	0.3343-13	0.6625-13	0.4069-13	0.3298-13	0.3274-13	0.4111-13
70	0.1711-13	0.3995-13	0.2749-13	0.4822-13	0.3605-13	0.3490-13	0.3160-13	0.4635-13
80	0.1514-13	0.4922-13	0.3191-13	0.4825-13	0.2867-13	0.3264-13	0.3416-13	0.3465-13
90	0.1521-13	0.5414-13	0.2129-13	0.5062-13	0.2868-13	0.2521-13	0.3335-13	0.3148-13
100	0.1294-13	0.2814-13	0.2105-13	0.4158-13	0.3608-13	0.2336-13	0.2131-13	0.3318-13
110	0.9133-14	0.2995-13	0.1897-13	0.3962-13	0.2606-13	0.1734-13	0.2100-13	0.2750-13
120	0.6591-14	0.2006-13	0.1376-13	0.3166-13	0.2162-13	0.1713-13	0.1843-13	0.1652-13
130	0.6516-14	0.1982-13	0.1967-13	0.2698-13	0.1927-13	0.1546-13	0.1265-13	0.1311-13
140	0.4228-14	0.1607-13	0.9407-14	0.2747-13	0.1302-13	0.1189-13	0.1137-13	0.1367-13
150	0.3163-14	0.1279-13	0.1245-13	0.2153-13	0.1038-13	0.1829-13	0.7176-14	0.9263-14
160	0.3668-14	0.1072-13	0.9068-14	0.1269-13	0.8464-14	0.5490-14	0.6895-14	0.8015-14
170	0.1990-14	0.5938-14	0.2794-14	0.7959-14	0.4975-14	0.3388-14	0.2168-14	0.4317-14
180	0.3009-15	0.1102-14	0.5746-15	0.1304-14	0.7574-15	0.8988-15	0.9027-15	0.1023-14

$0.3489\text{-}13 = 3.489 \times 10^{-14}$

43
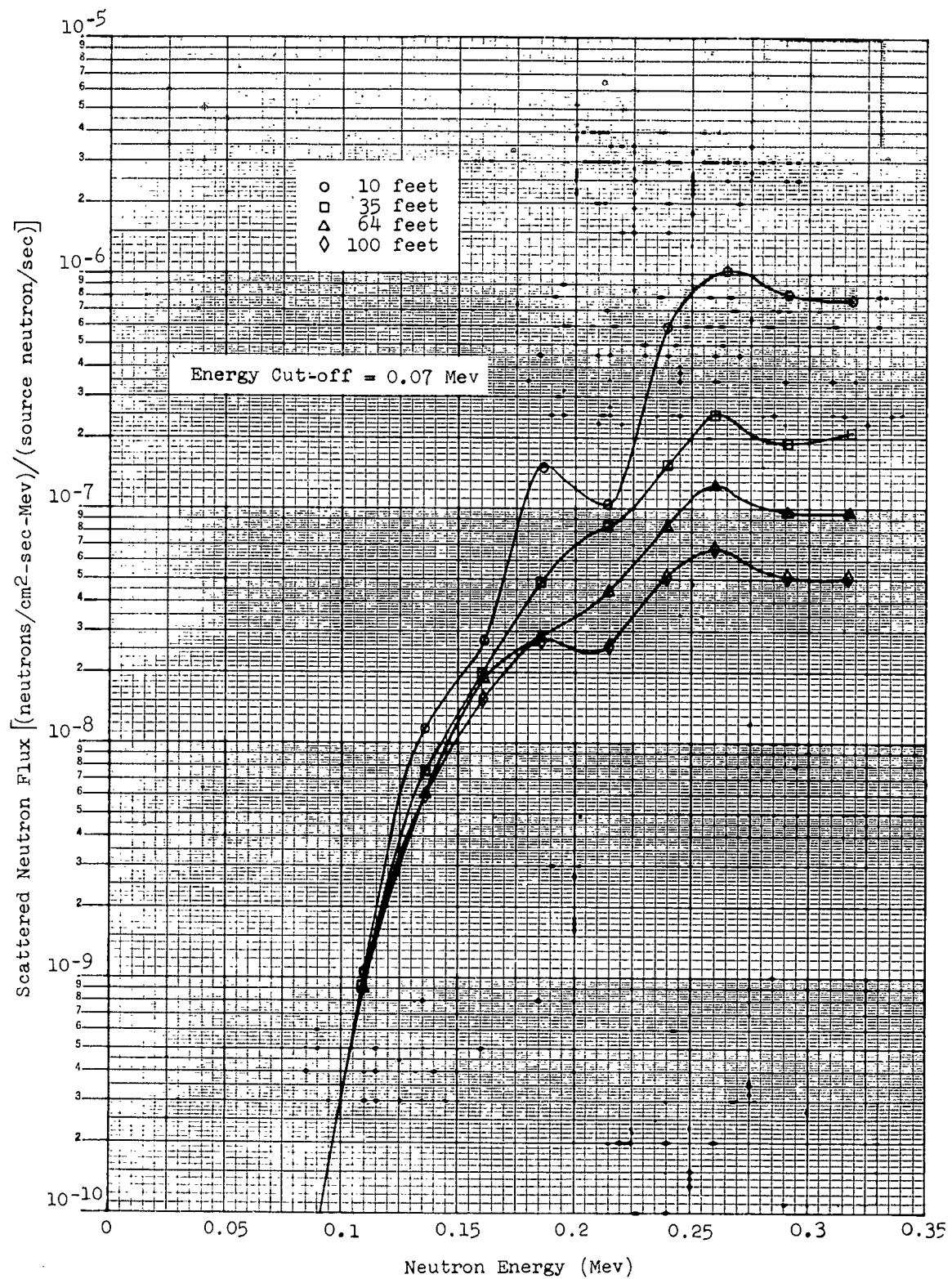
FIGURE 2. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 0.33 Mev

44
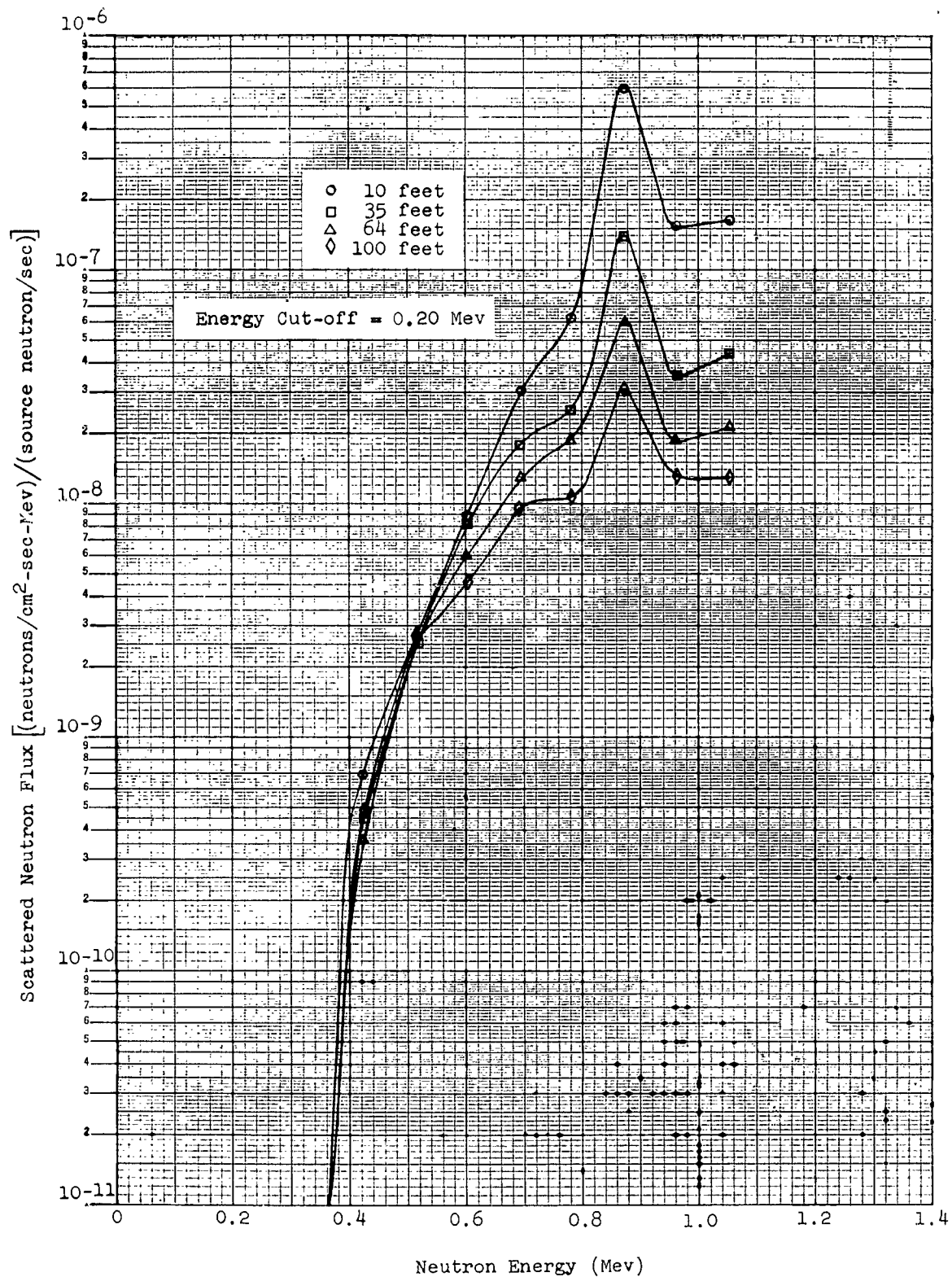
FIGURE 3. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 1.1 Mev

45
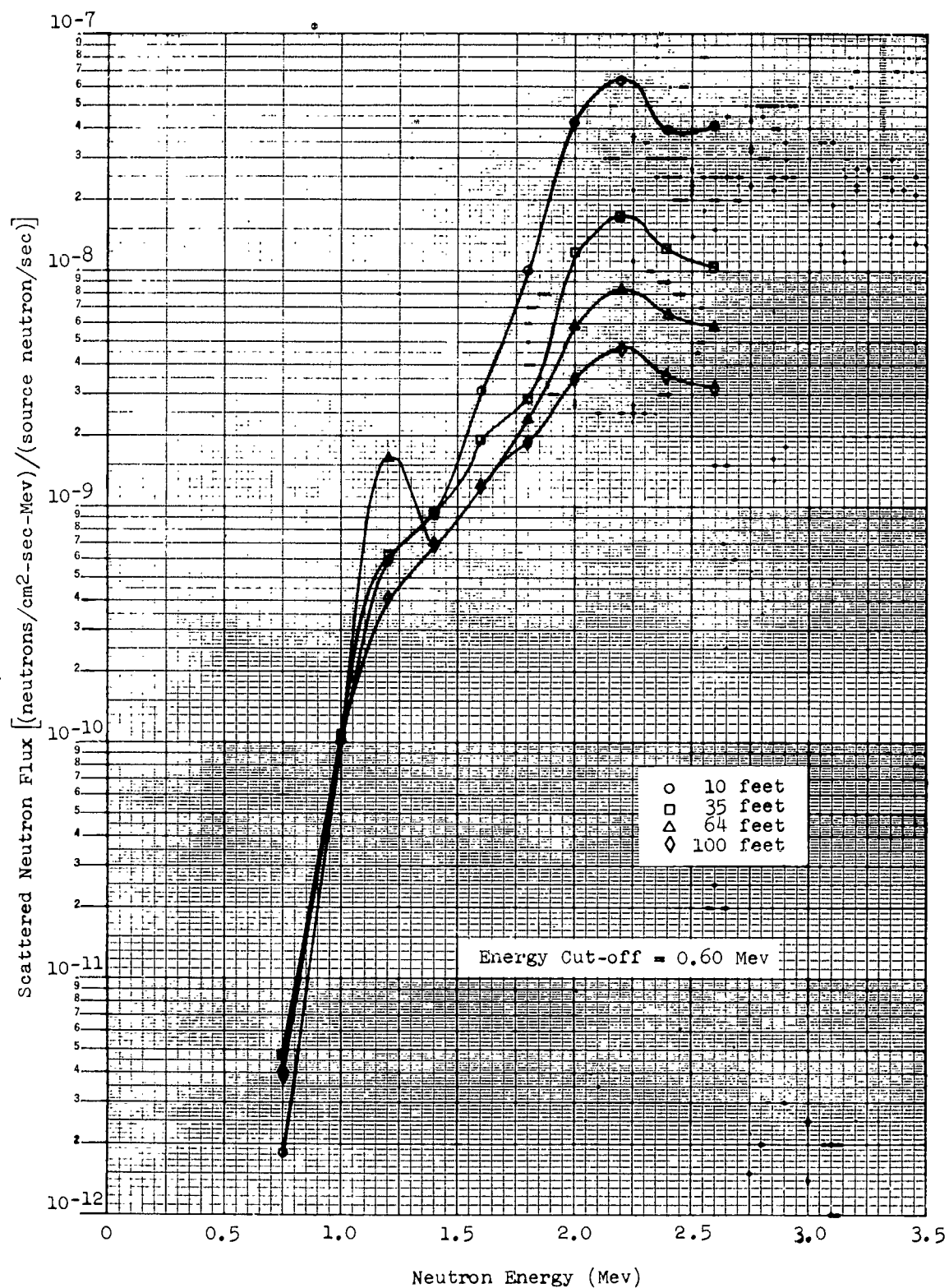
FIGURE 4. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 2.7 Mev

46
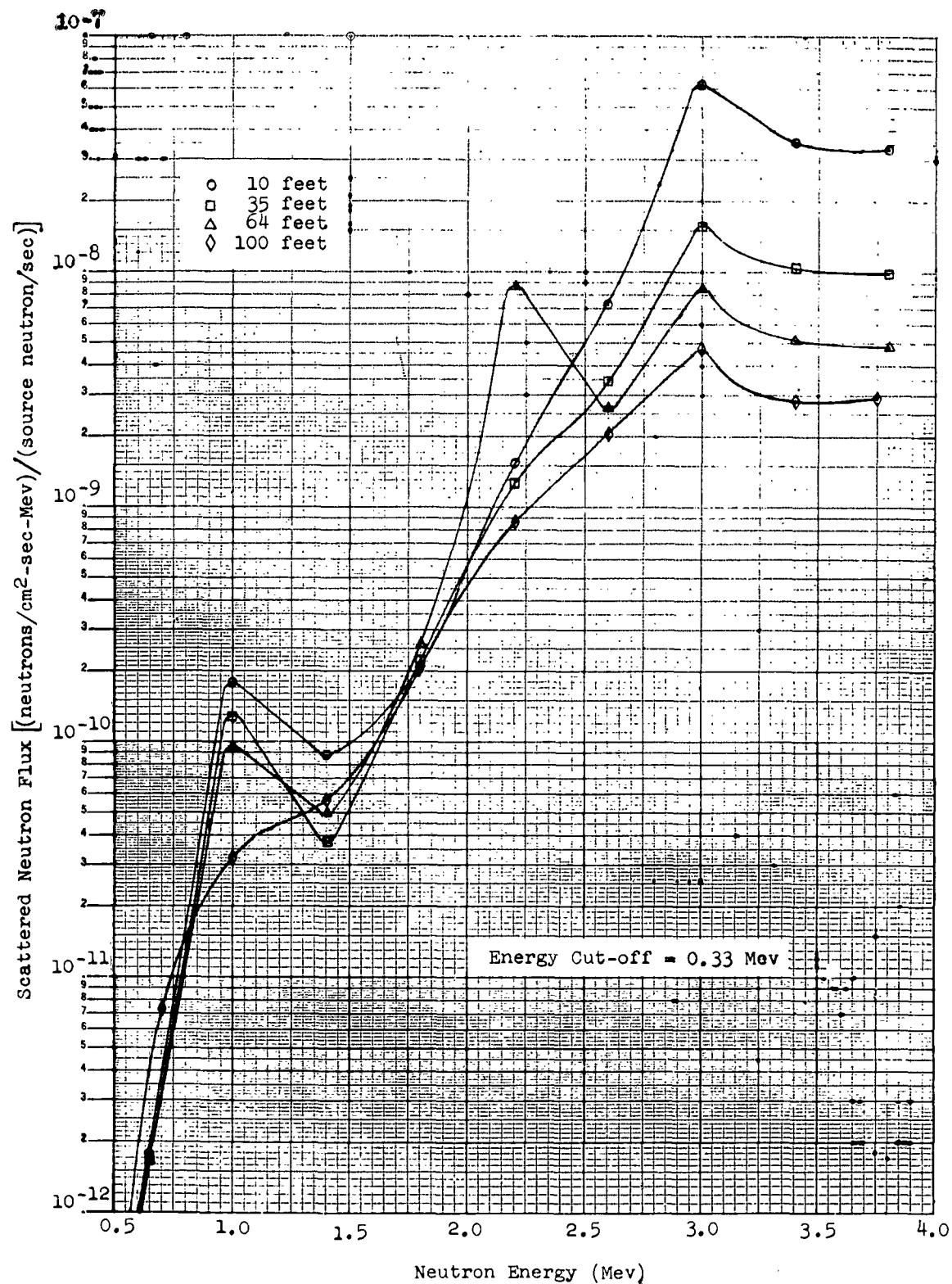
FIGURE 5.   ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 4.0 Mev

47
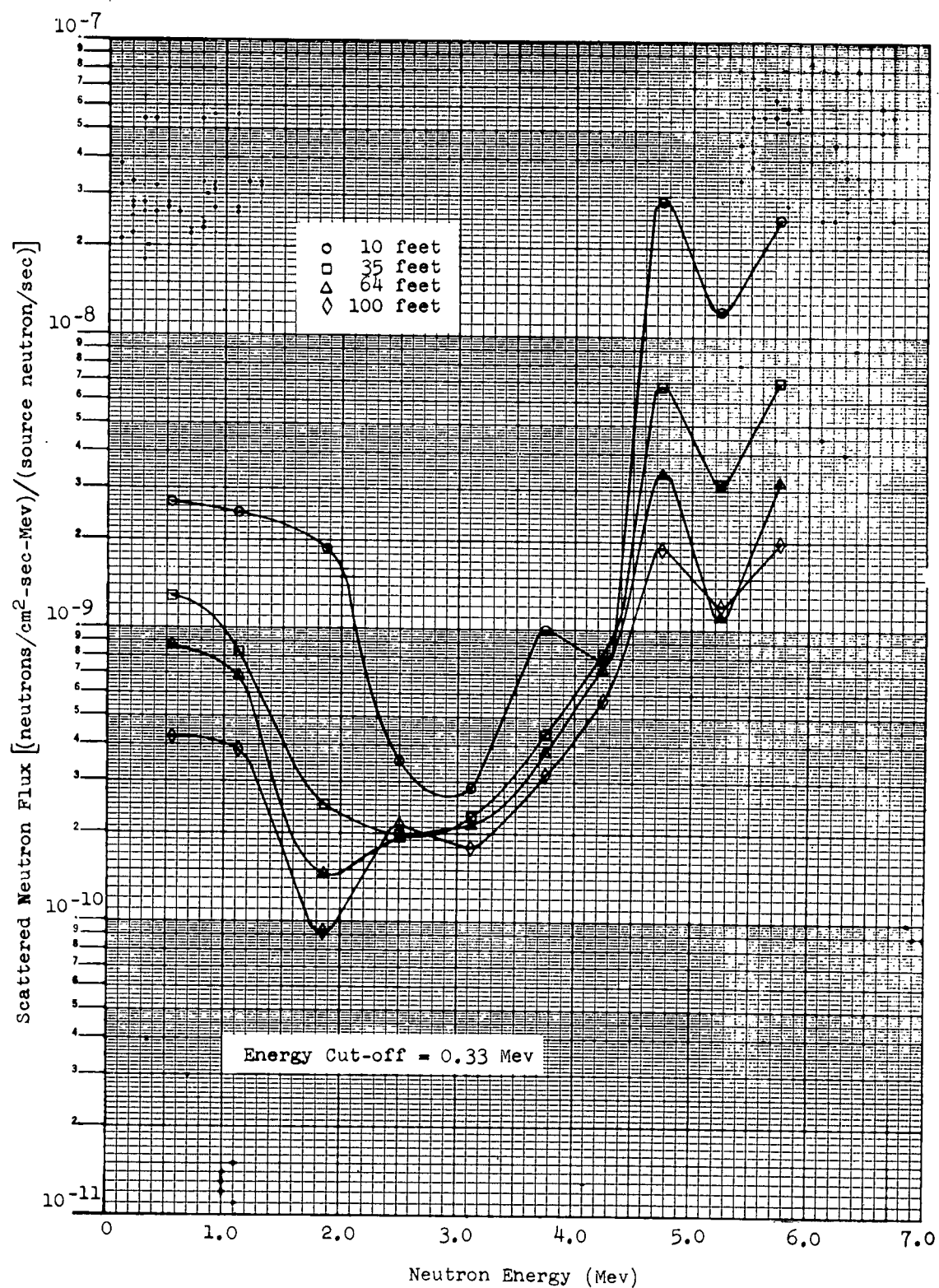
FIGURE 6. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 6.0 Mev

48
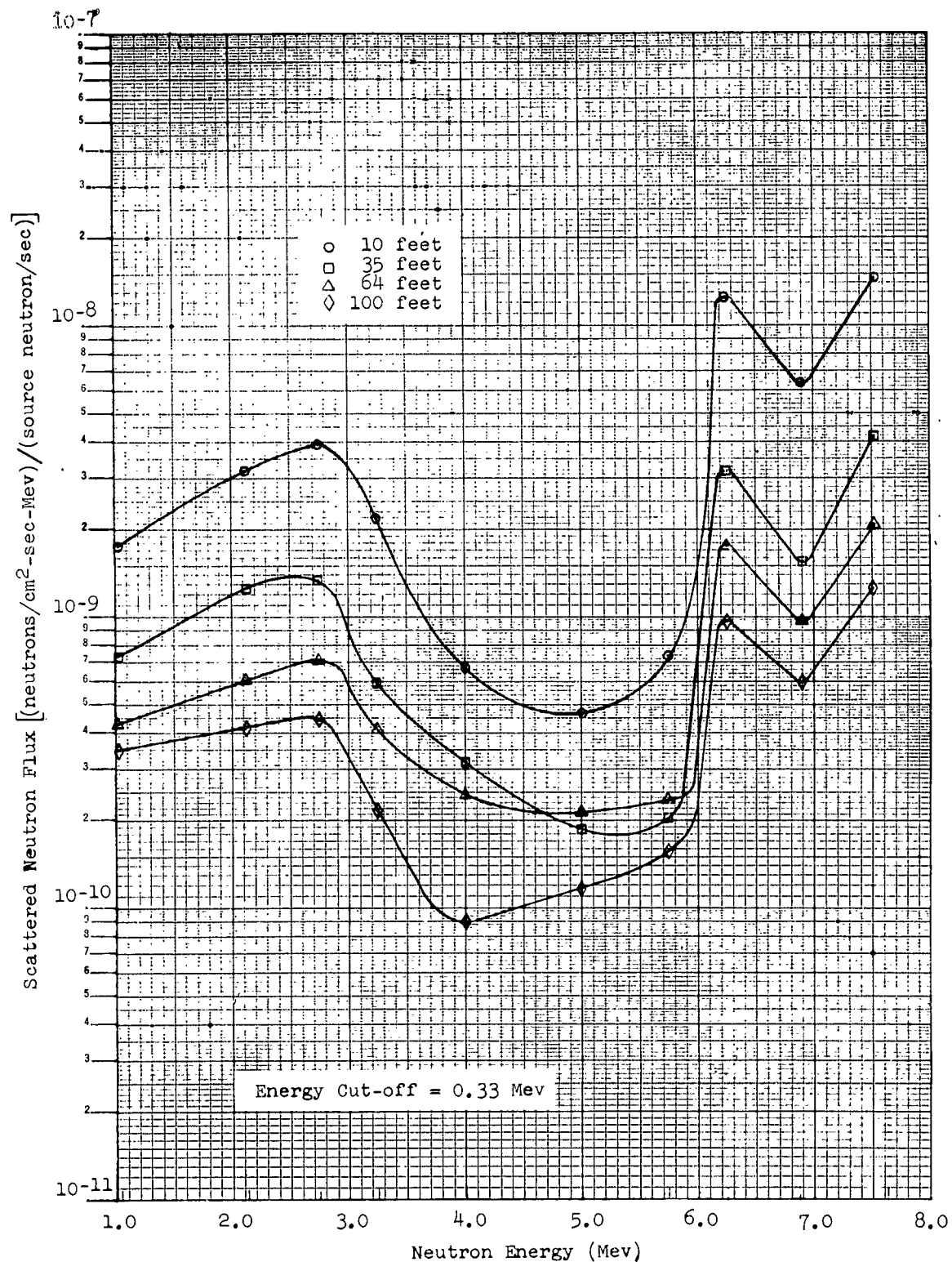
FIGURE 7. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 8.0 Mev

49
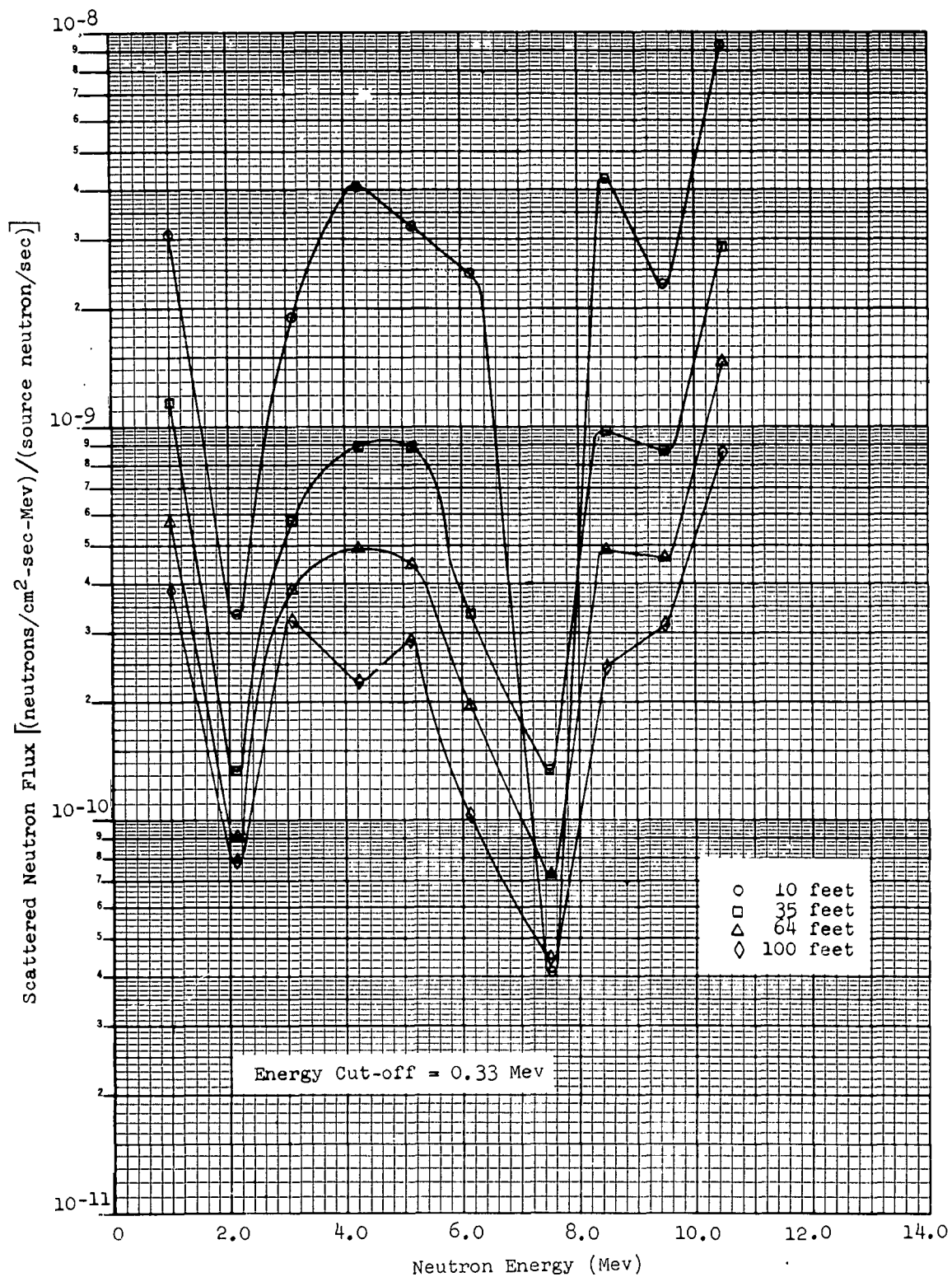
FIGURE 8. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 10.9 Mev

50
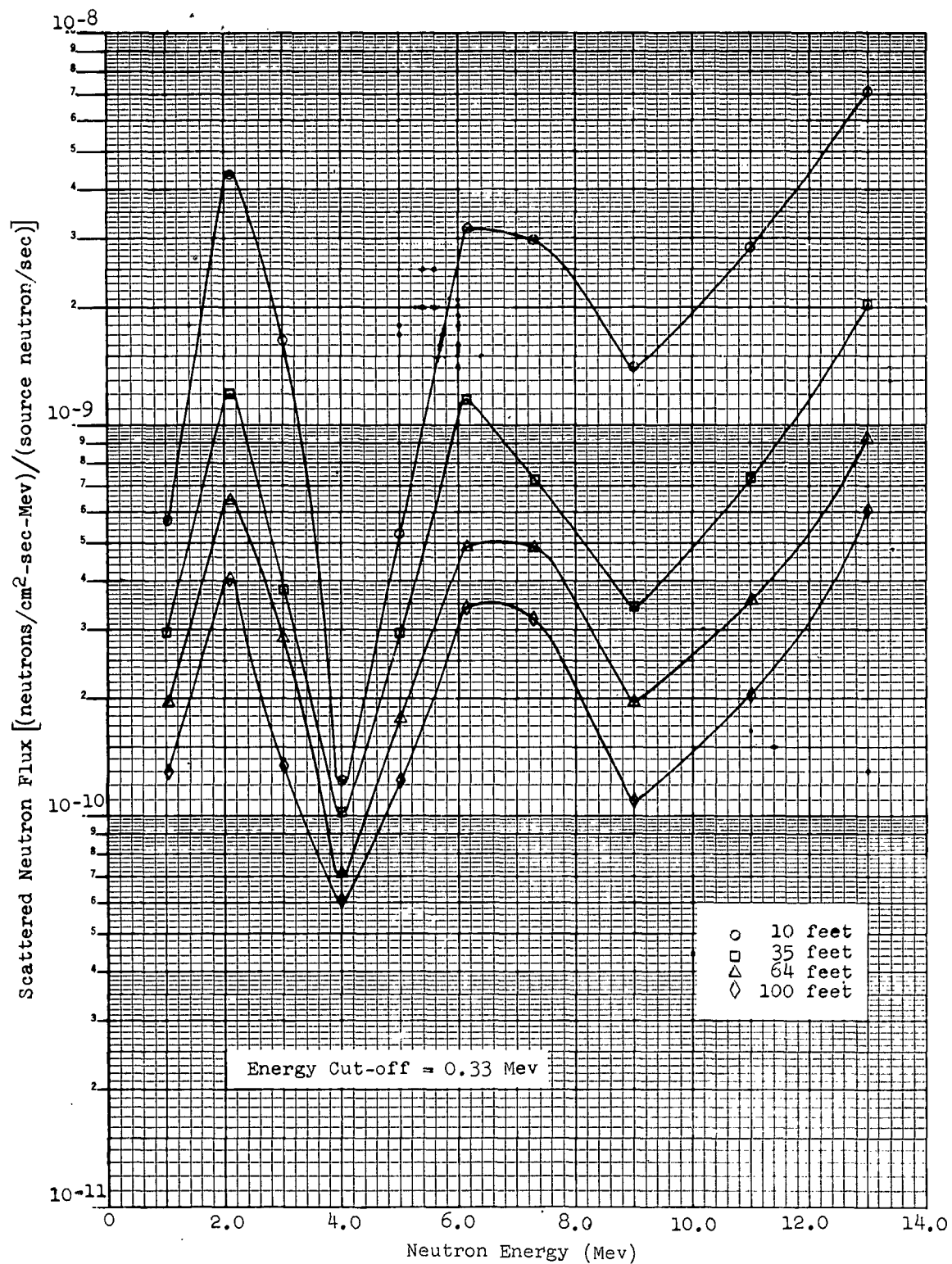
FIGURE 9. ENERGY SPECTRA OF TOTAL SCATTERED NEUTRON FLUX
Initial Energy 14.0 Mev

51
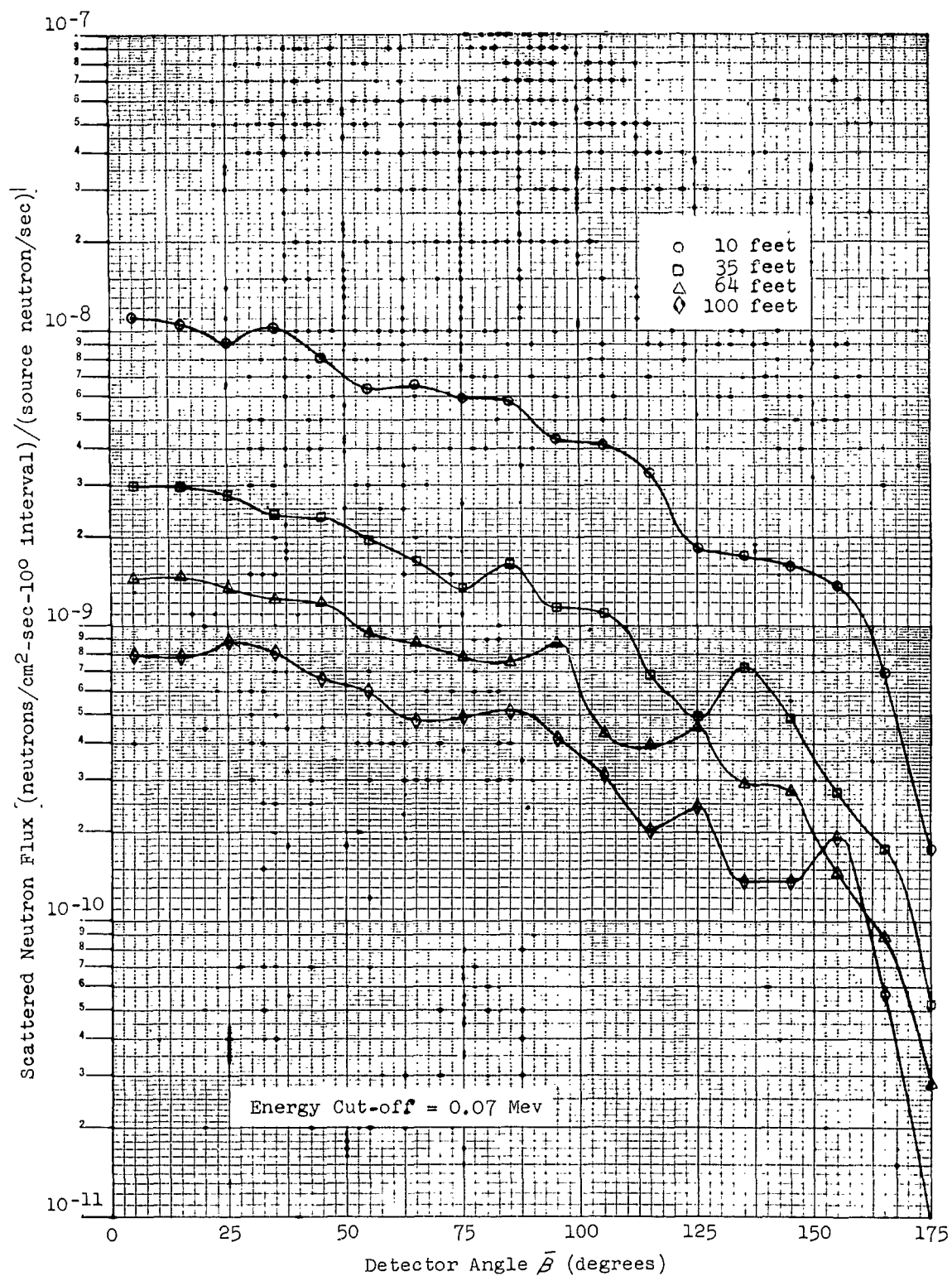
FIGURE 10. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 0.33 Mev

52
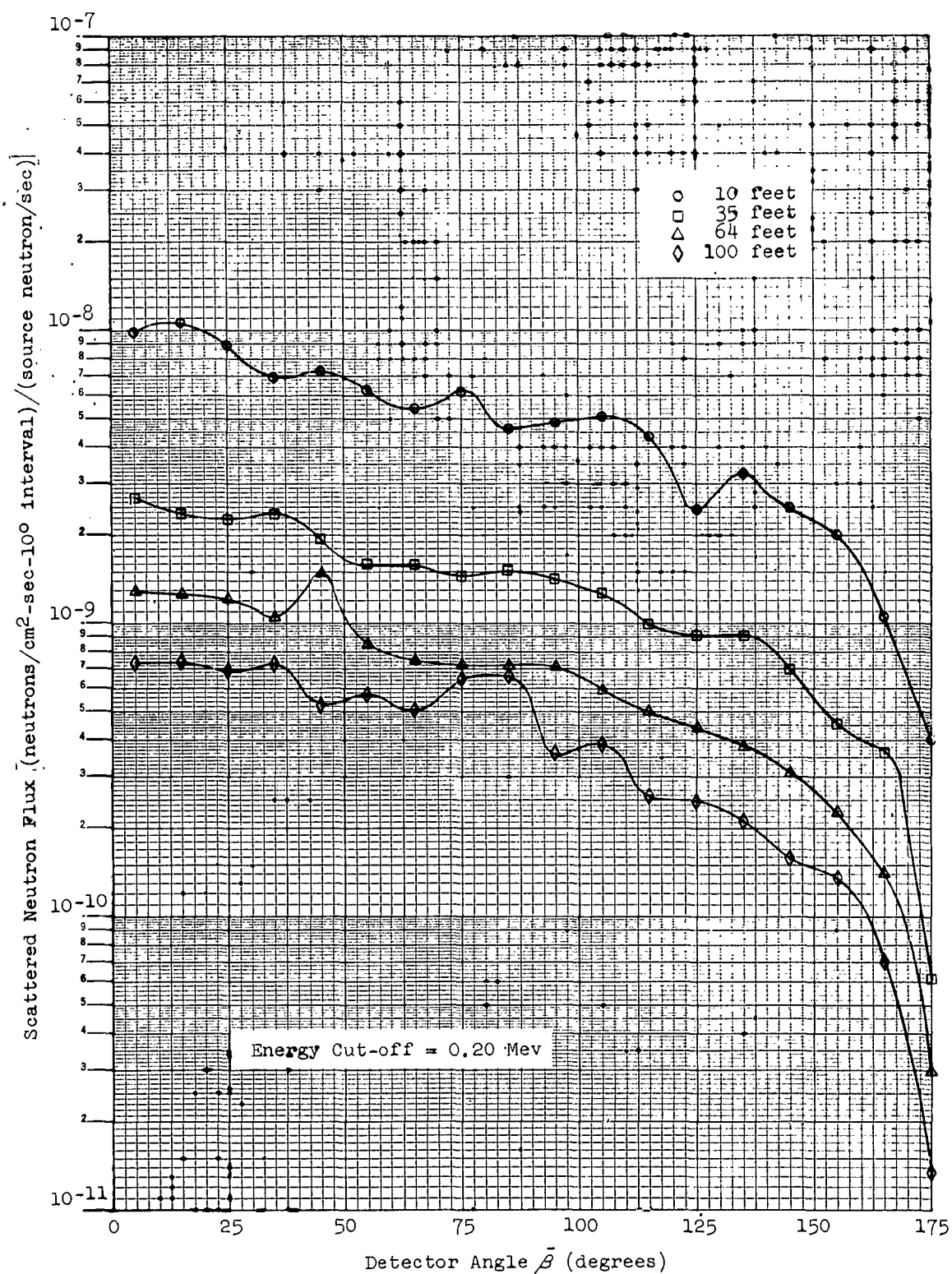
FIGURE 11.   TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 1.1 Mev

53
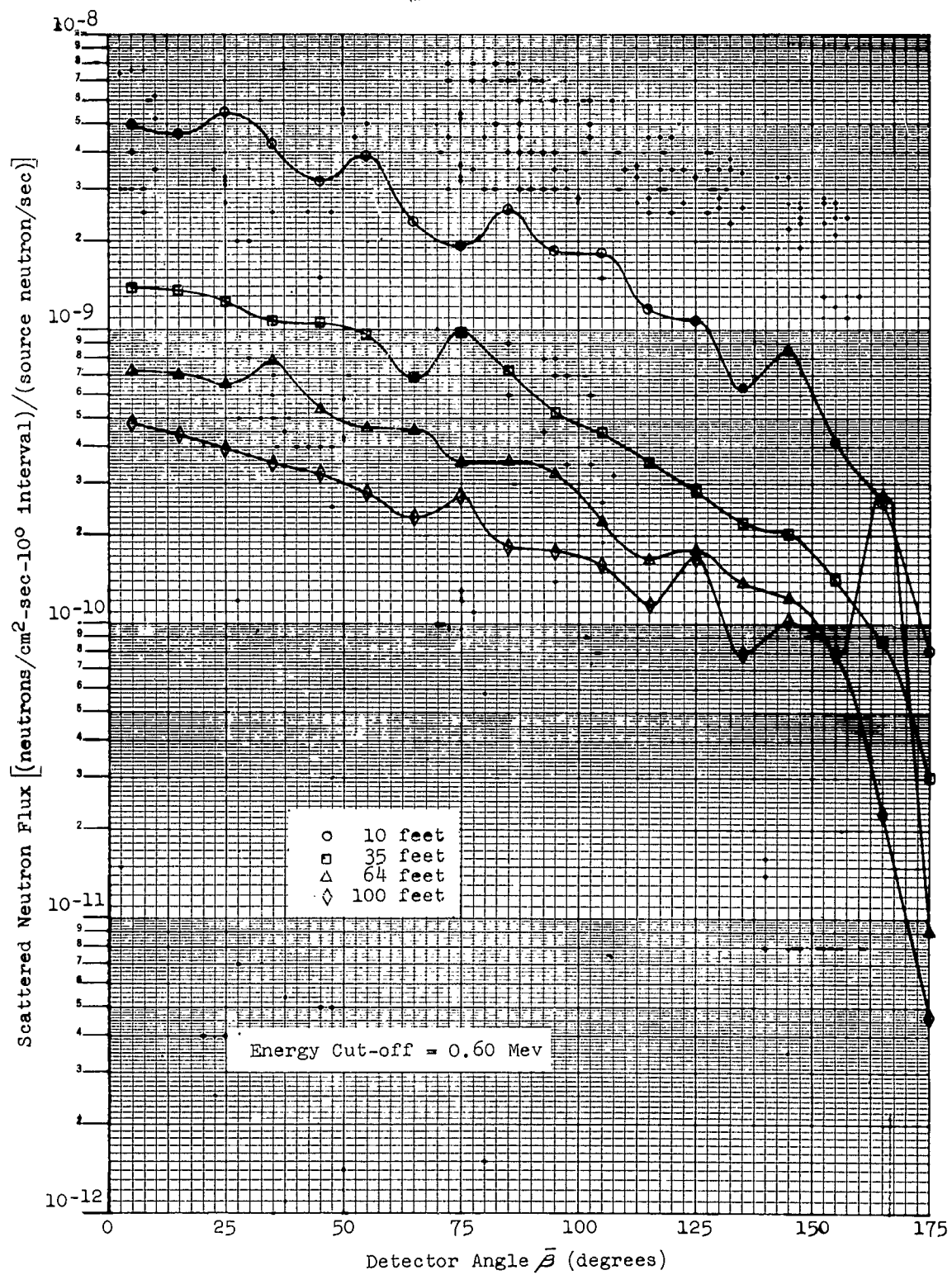
FIGURE 12.   TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 2.7 Mev

54
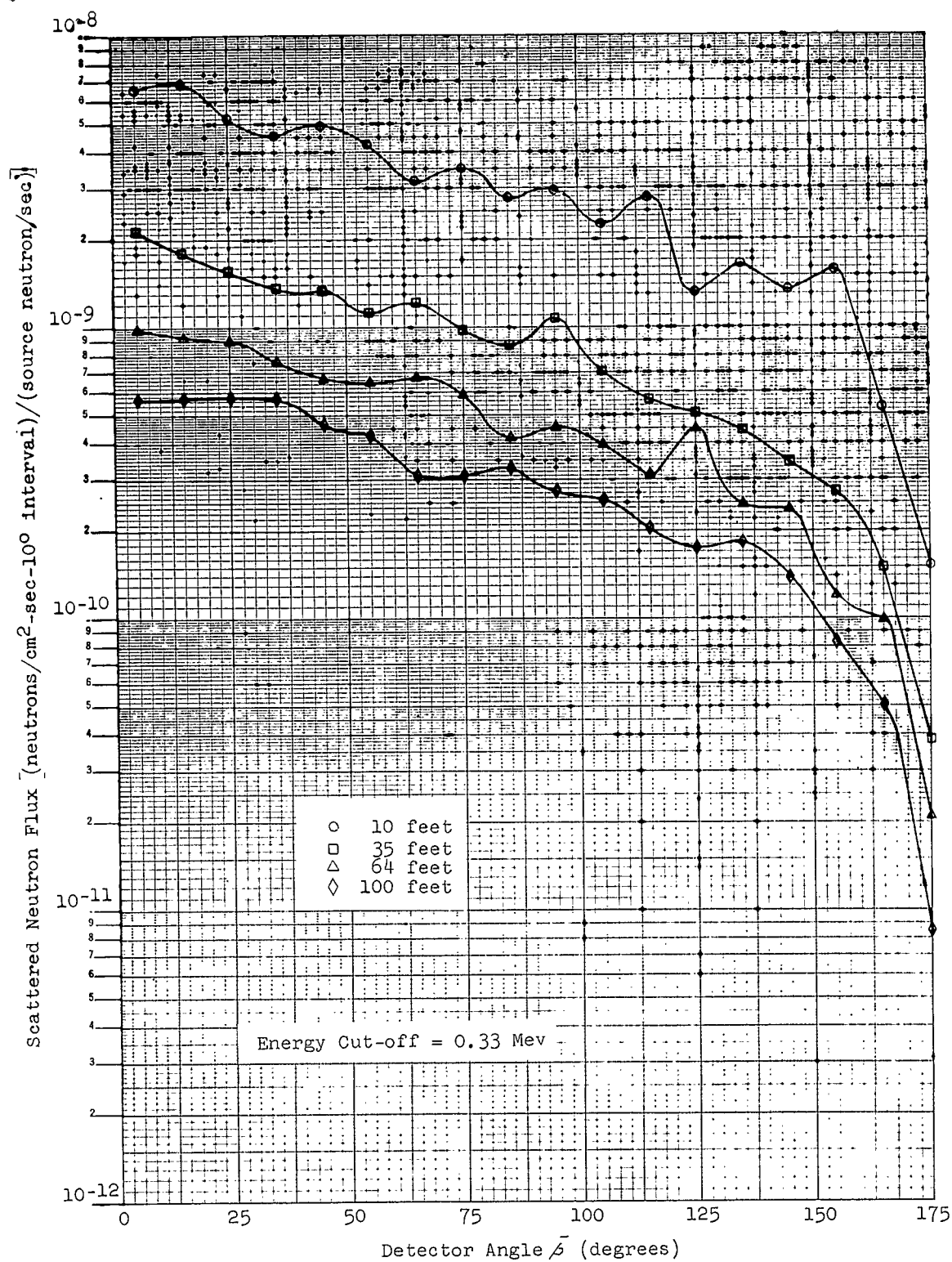
FIGURE 13. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 4.0 Mev

55
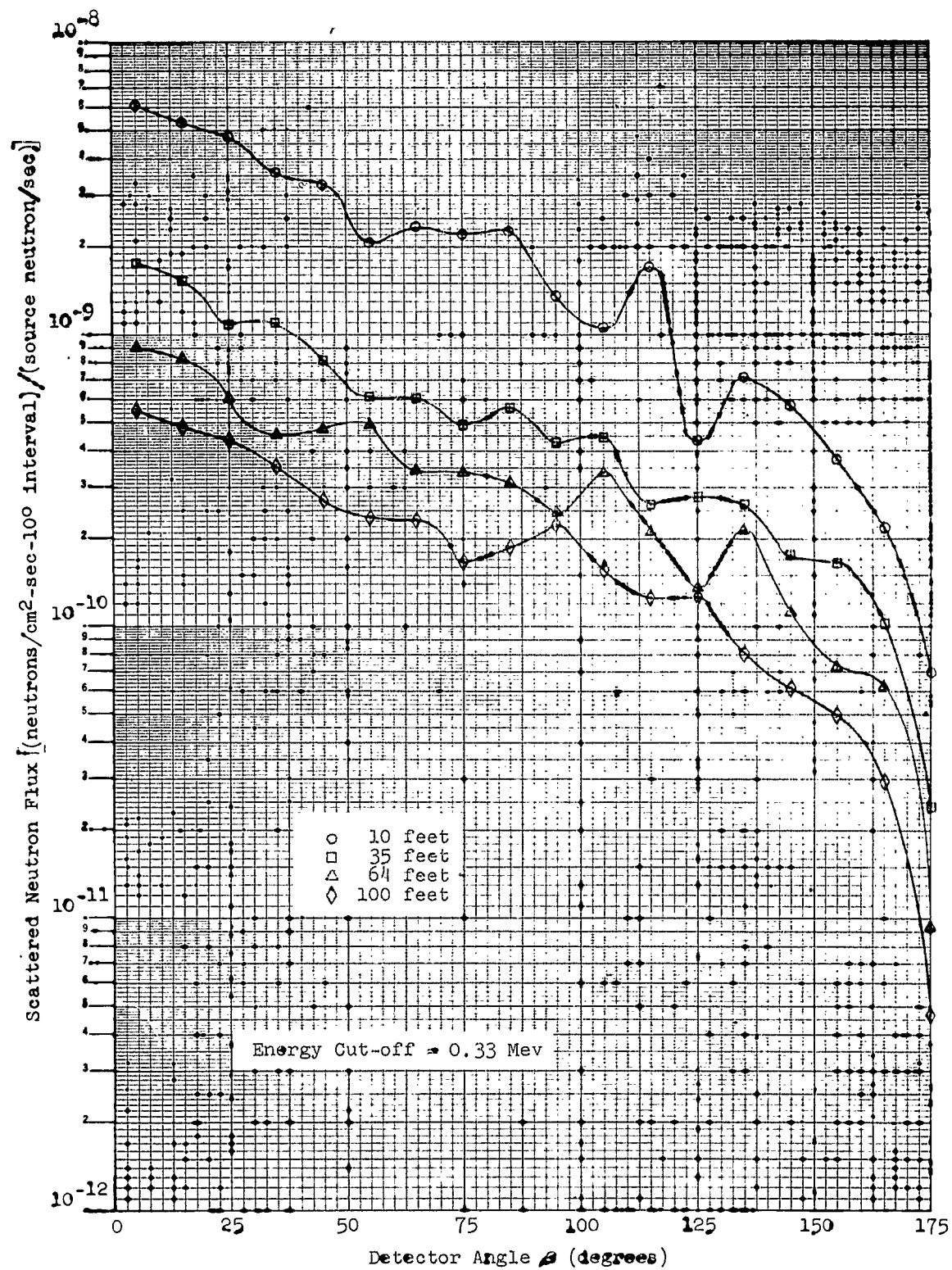
FIGURE 14. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 6.0 Mev

56
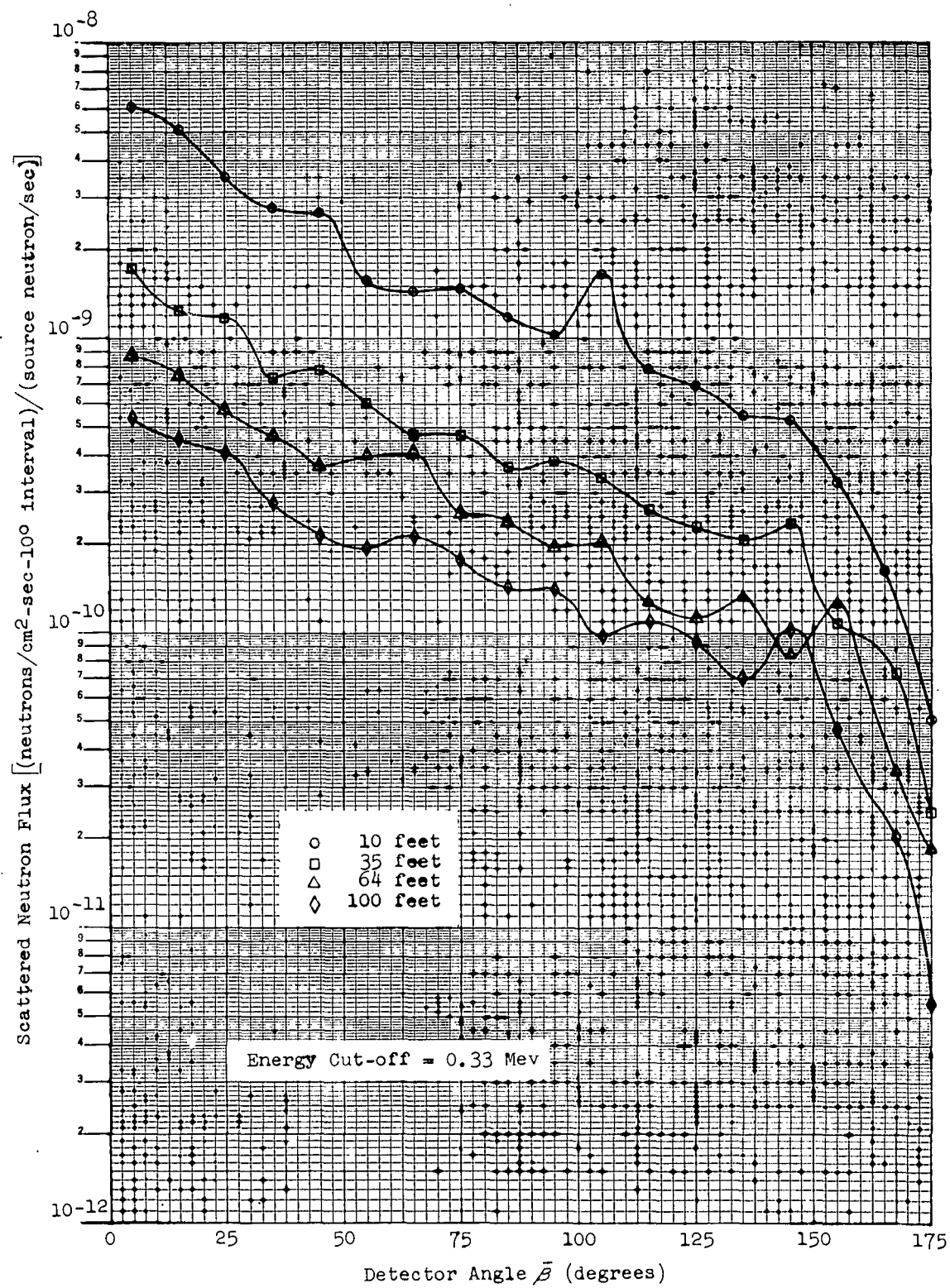
FIGURE 15.   TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 8.0 Mev

57
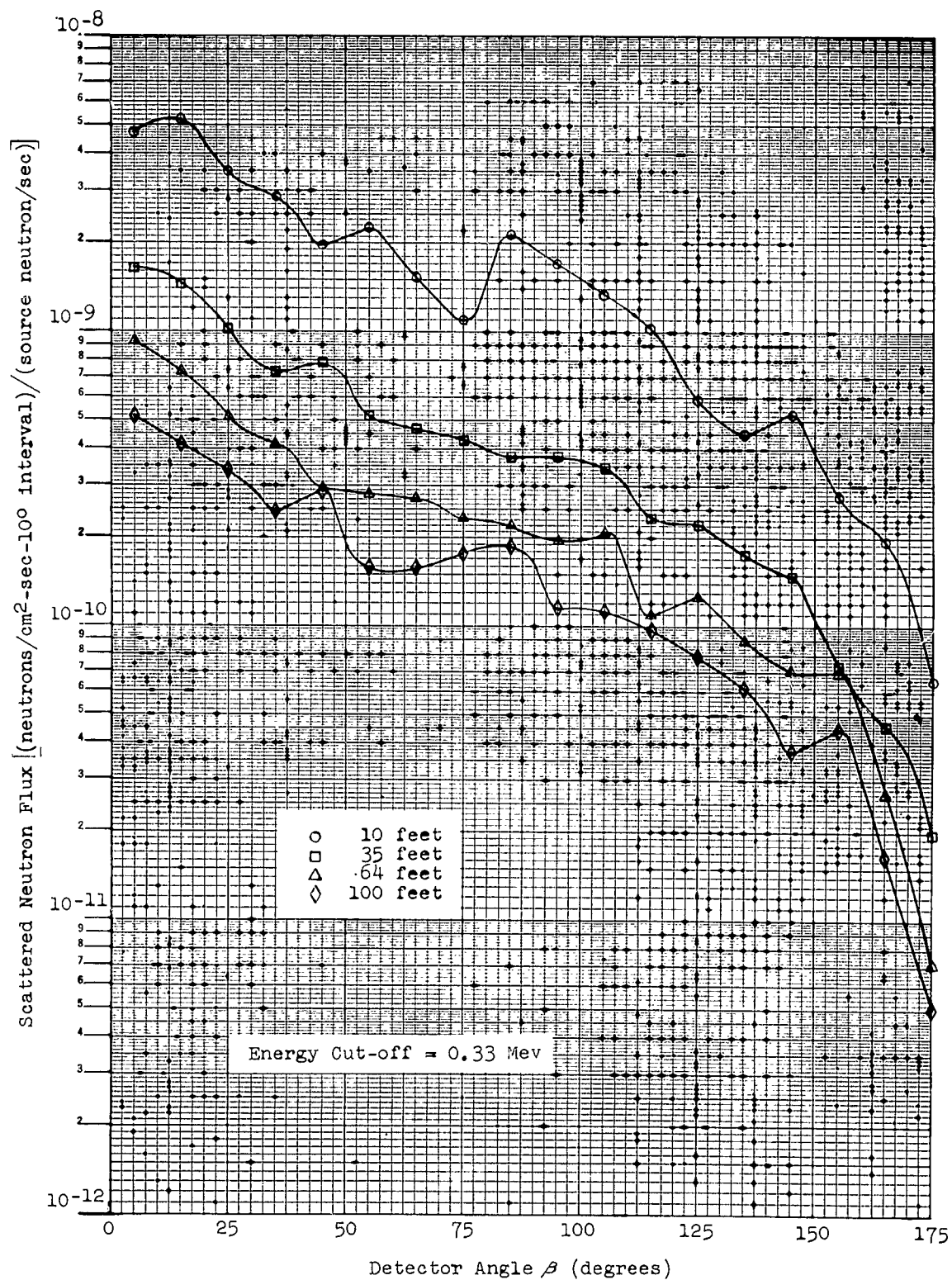
FIGURE 16.   TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR ANGLE
Initial Energy 10.9 Mev

58
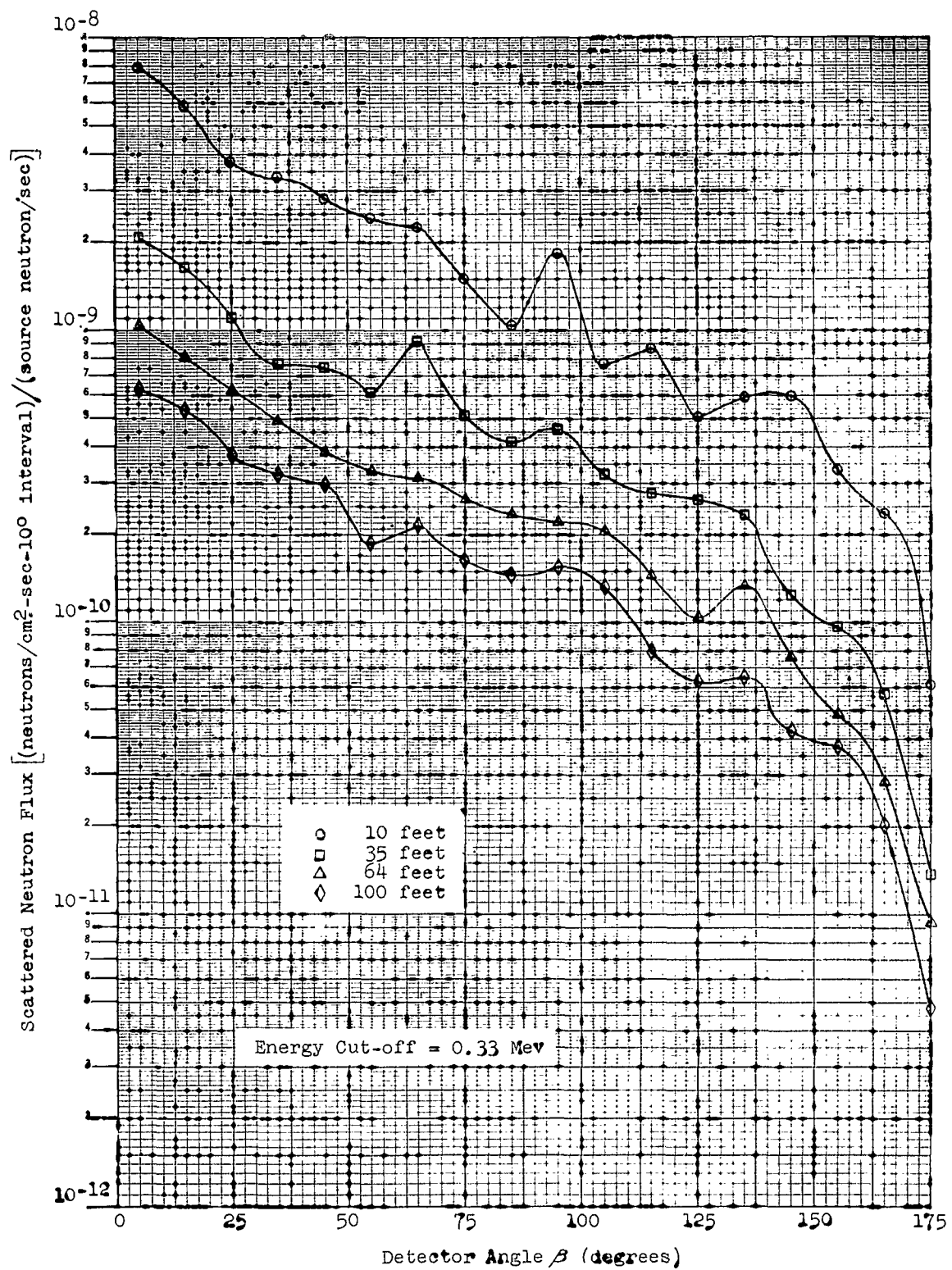
FIGURE 17. TOTAL SCATTERED NEUTRON FLUX VS. DETECTOR A: ...
Initial Energy 14.0 Mev
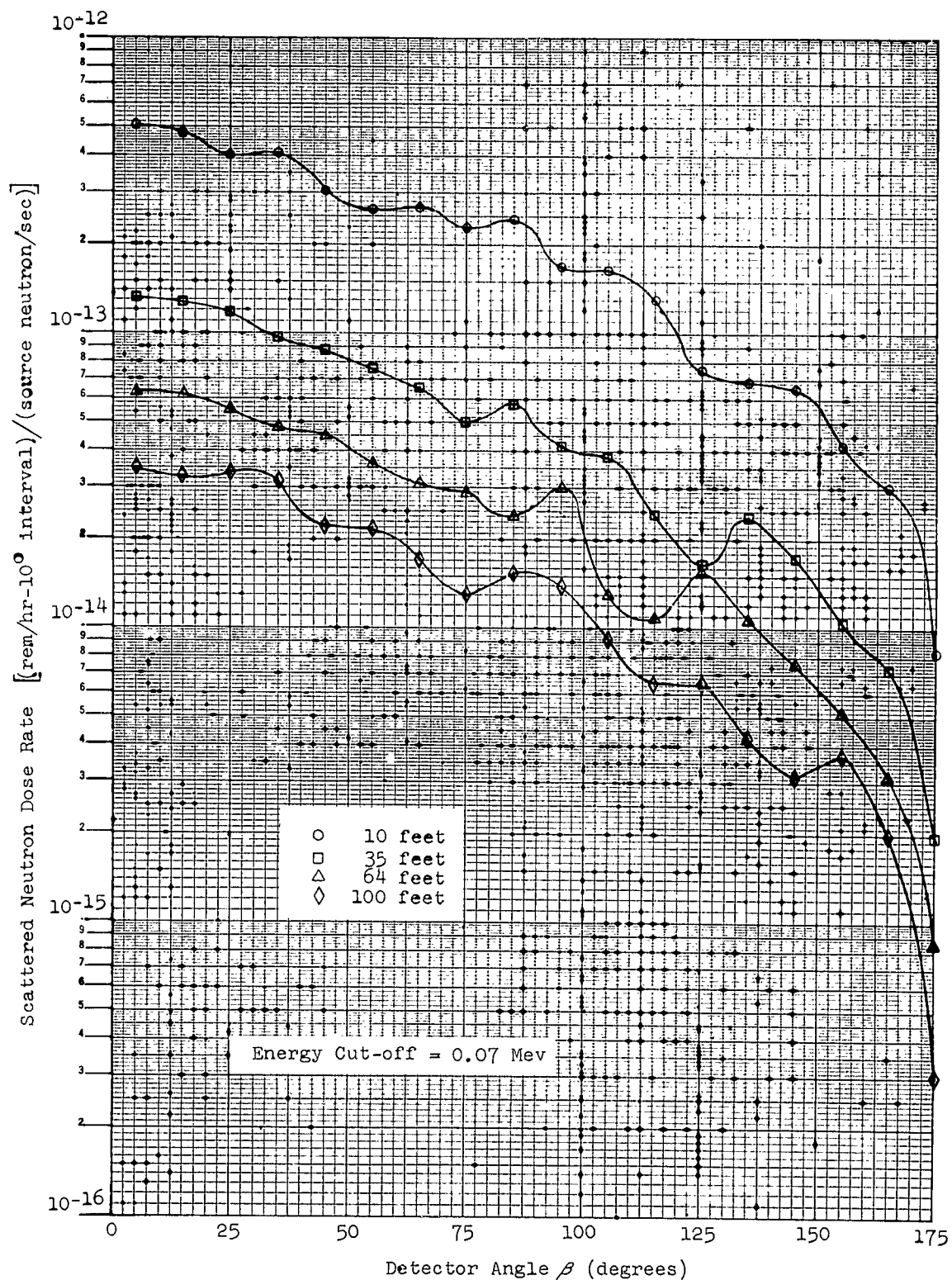
FIGURE 18. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 0.33 Mev

60
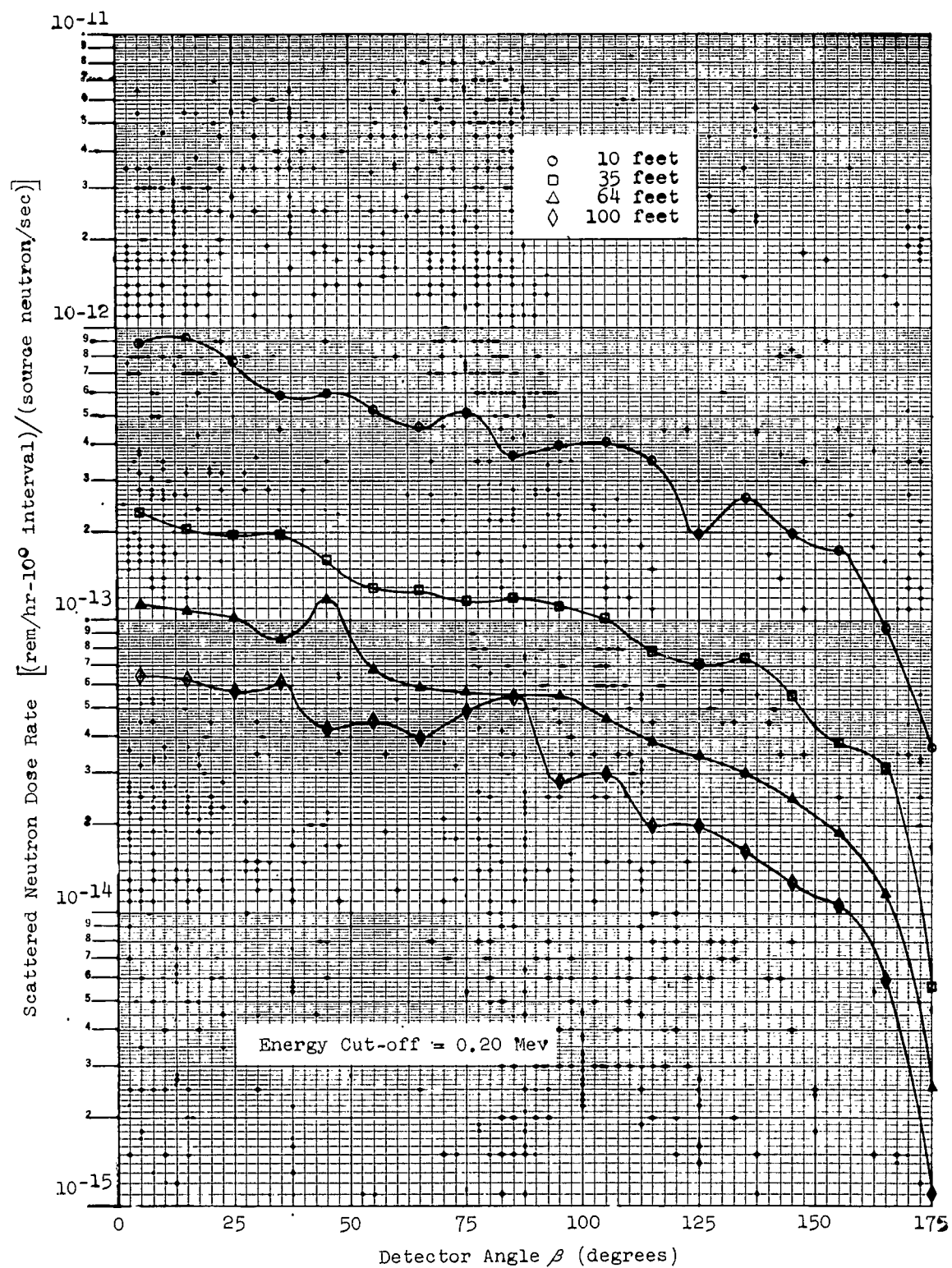
FIGURE 19.   TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 1.1 Mev

61
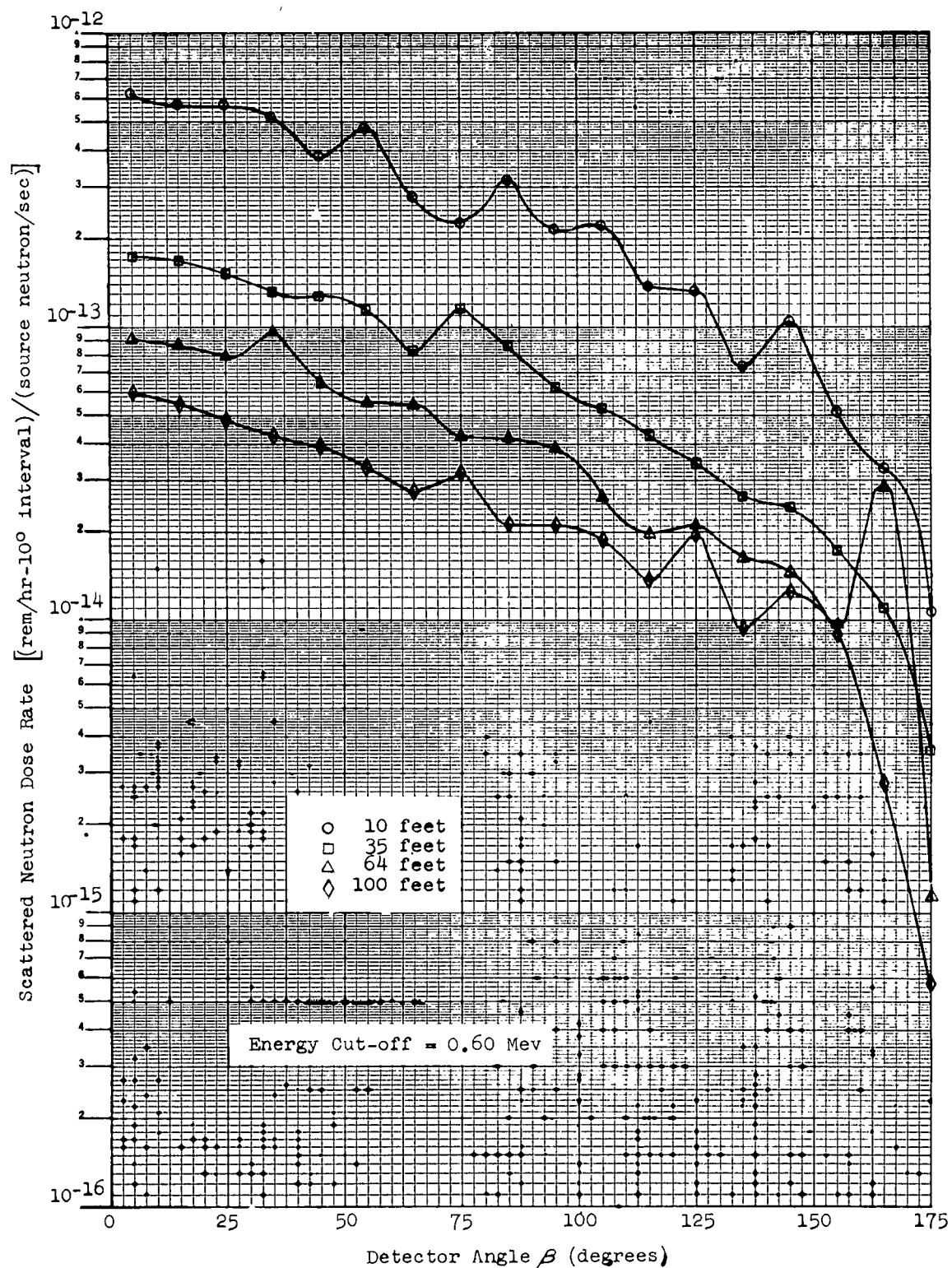
FIGURE 20. TOTAL SCATTERED NEUTRON DOSE RATE VS. D ECT'₌ ..:3LE
Initial Energy 2.7 Me₁

62
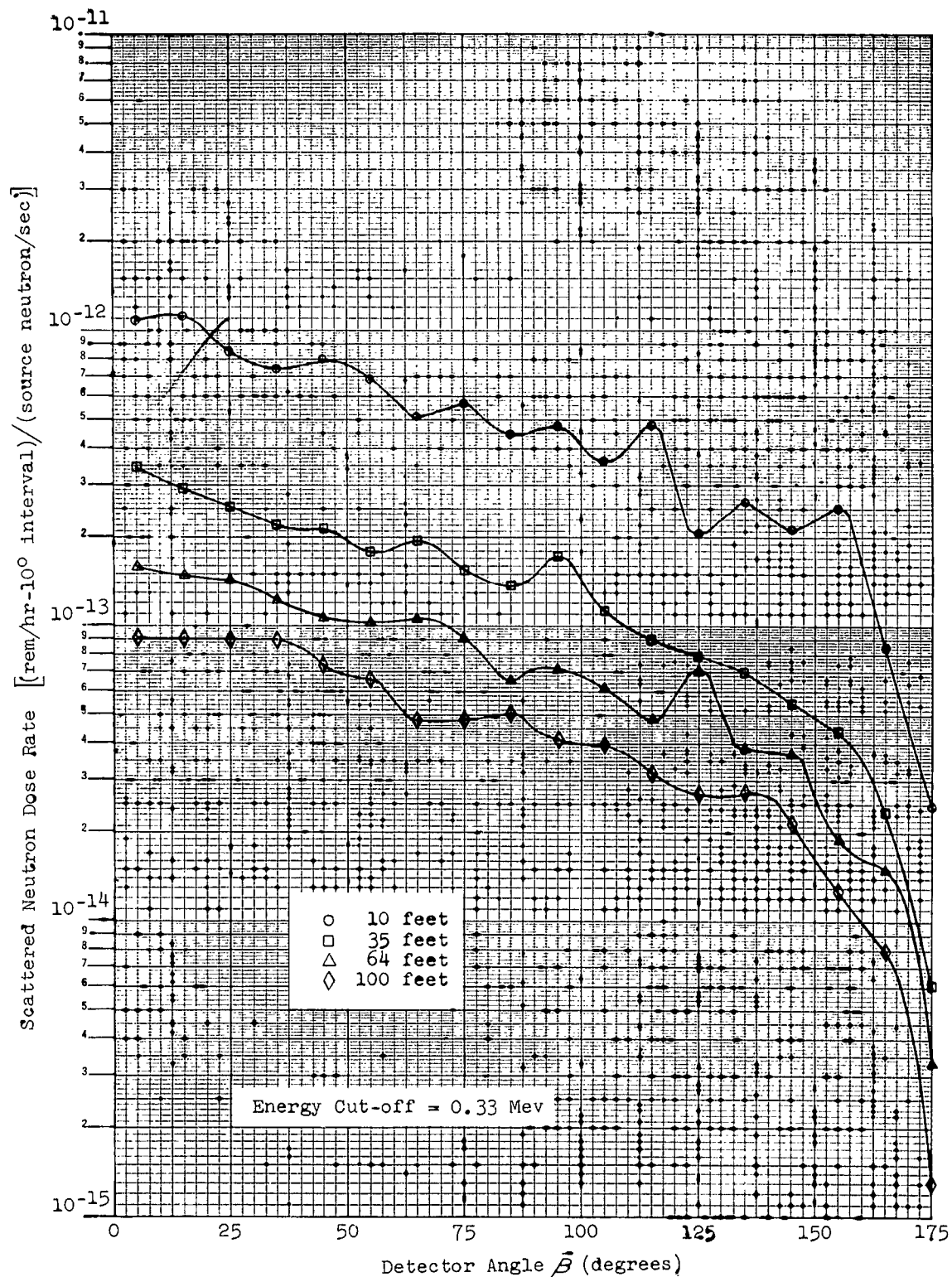
FIGURE 21.   TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 4.0 Mev

63
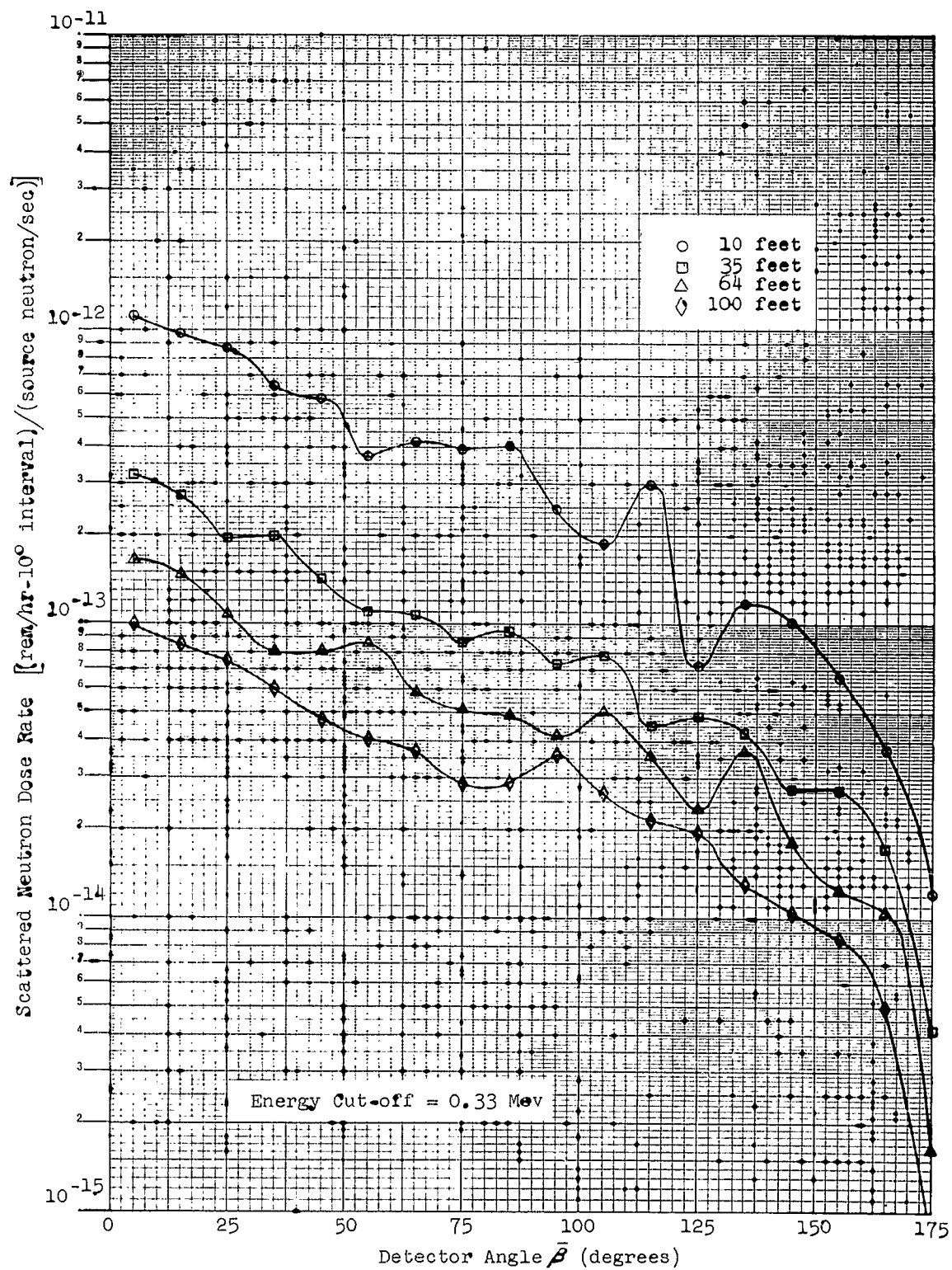
FIGURE 22. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 6.0 Mev

64
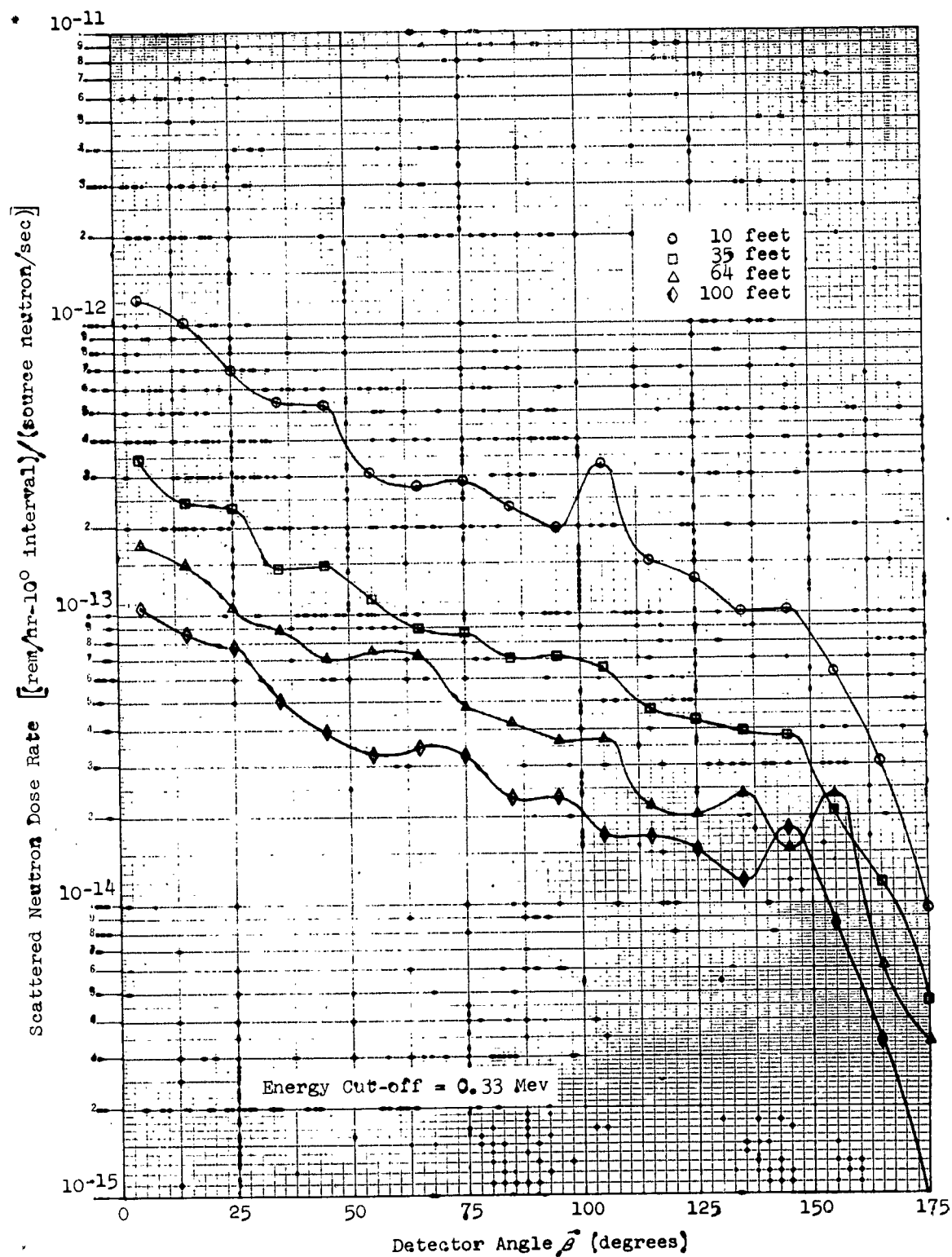
FIGURE 23. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 8.0 Mev

65
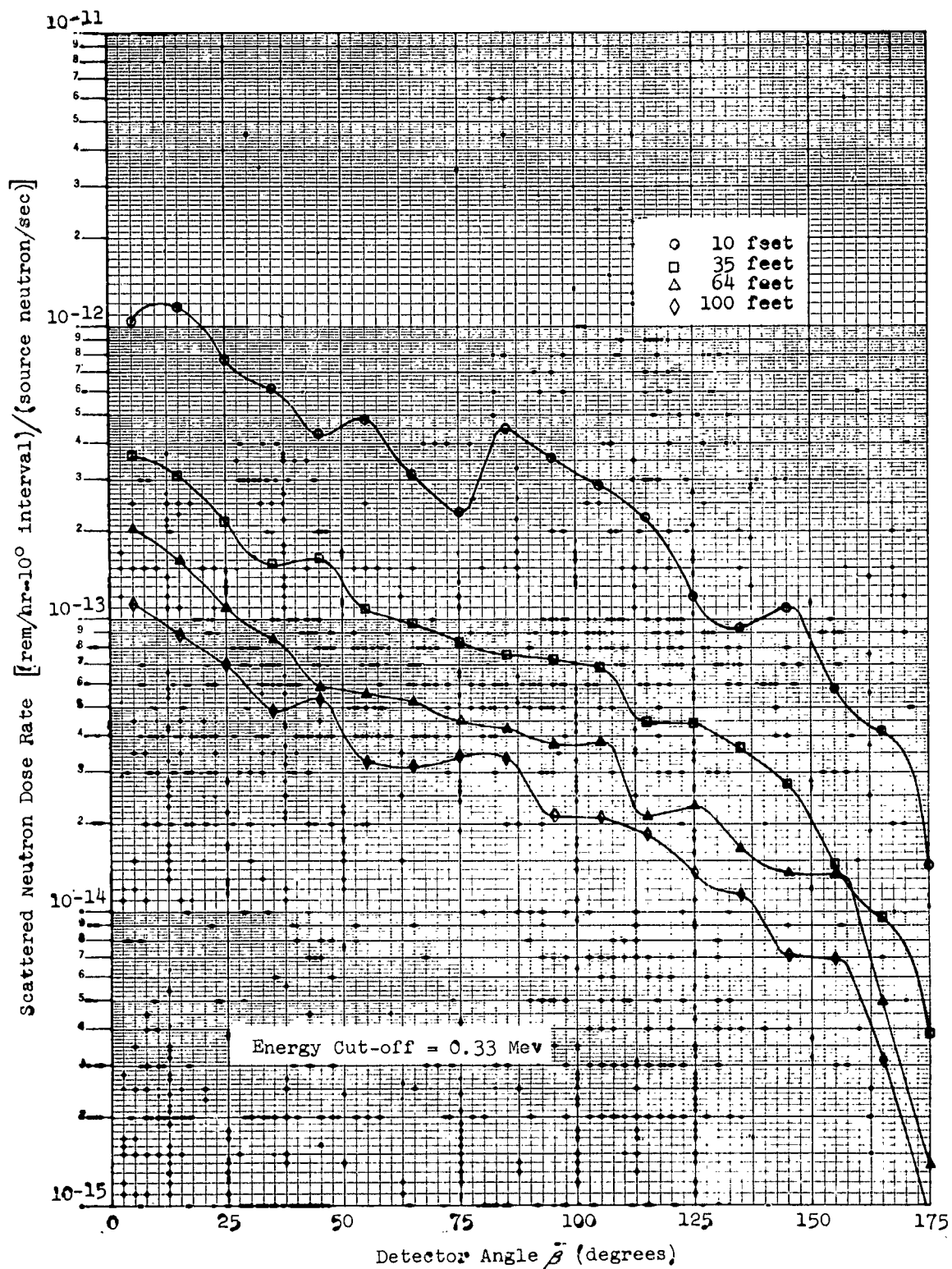
FIGURE 24. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 10.9 Mev

66
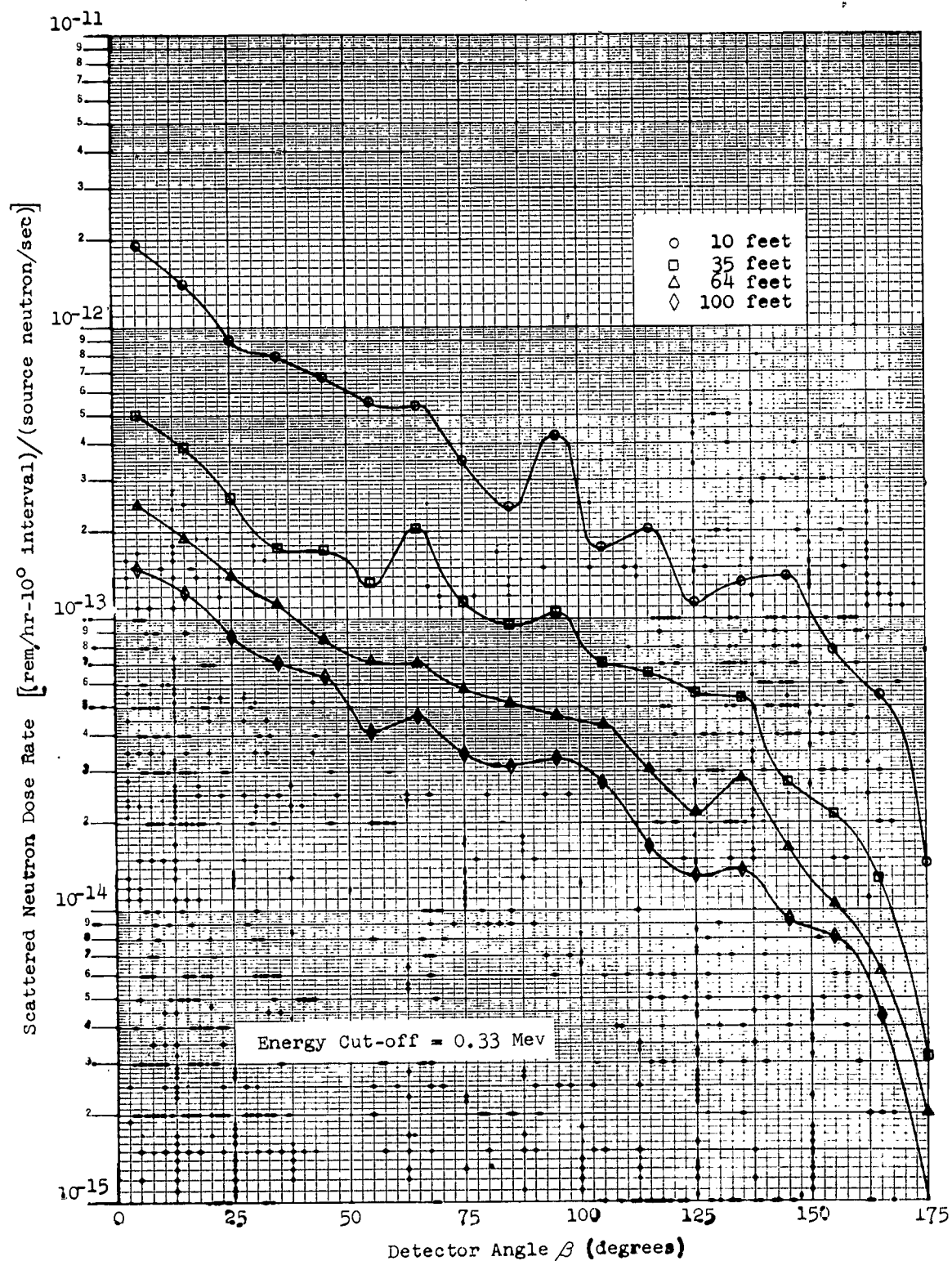
FIGURE 25. TOTAL SCATTERED NEUTRON DOSE RATE VS. DETECTOR ANGLE
Initial Energy 14.0 Mev

67
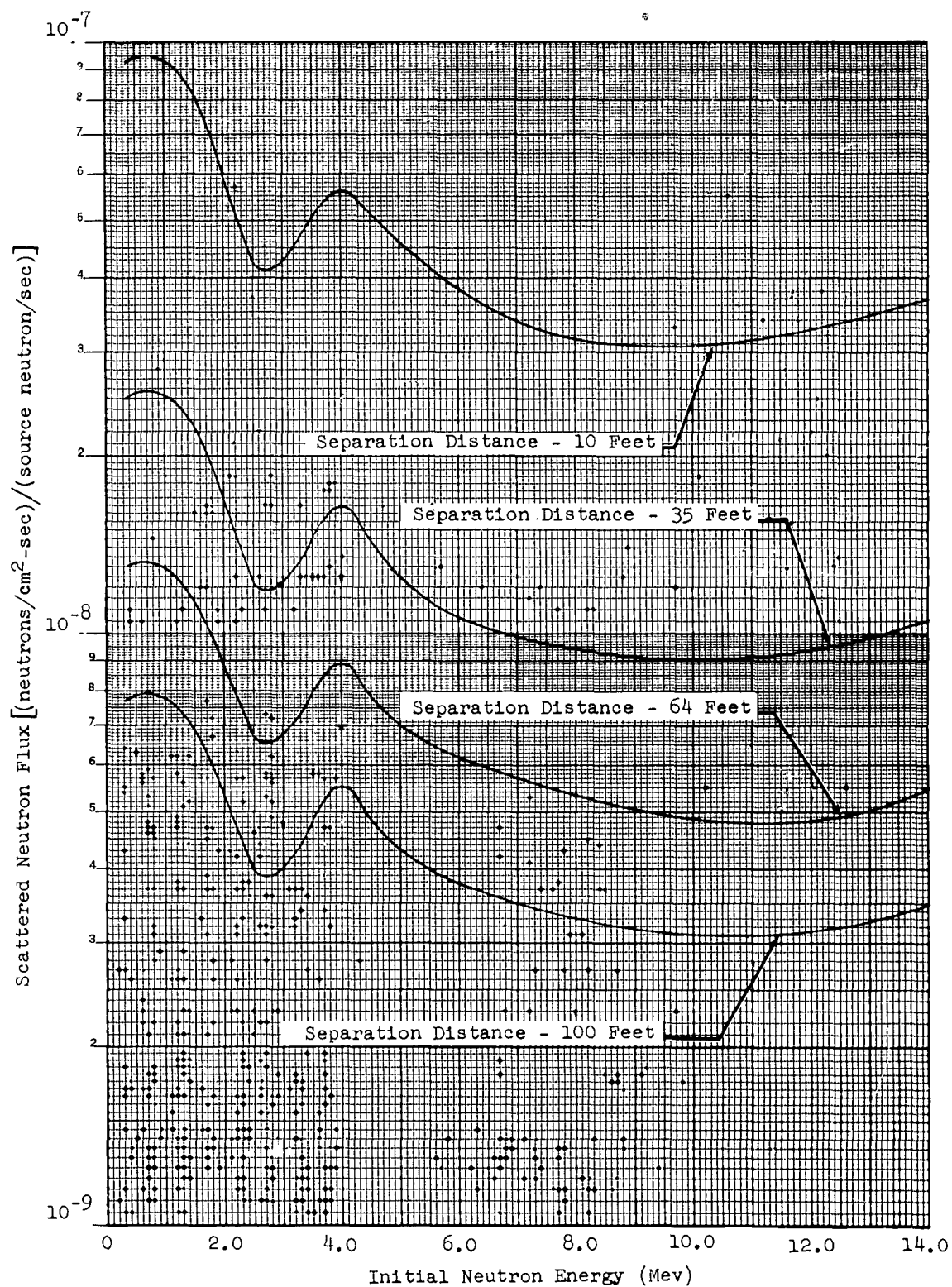
FIGURE 26. TOTAL SCATTERED NEUTRON FLUX VS. INITIAL ENERGY

68
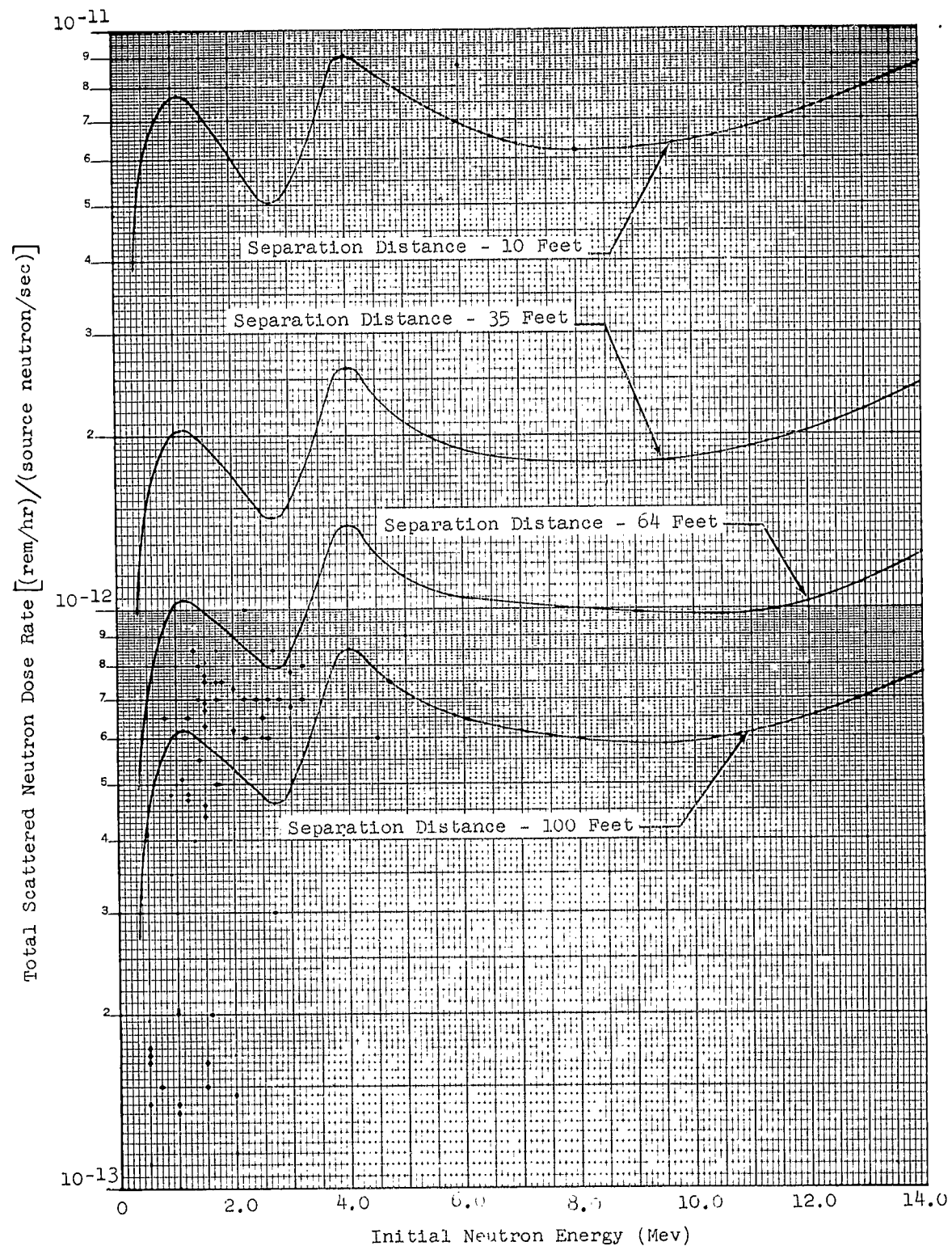
FIGURE 27. TOTAL SCATTERED NEUTRON DOSE RATE VS. INITIAL ENERGY

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APPENDIX

R-55 FORTRAN STATEMENTS

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'55 FORTRAN STATEMENTS

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 7171   BEG XTRA R55 JACK.GRIGSBY (CAROL DIFFEY-*3987)                    0001 .R55
EZE   EZEC1 EZED2 EZED3 EOF   EZED5 EZEB1AEZEB3 EZEB4 EZEB1AEZEB5
C       IBM 704 PROCEDURE R55                                            0002 R55
C       INTEGRATION OF MONTE CARLO CALCULATIONS OF FAST NEUTRON SCATTERING0003 R55
C       IN AIR FOR NON-ISOTROPIC SOURCES                                 0004 R55
C                                                                        0005 R55
        DIMENSION  Z(8),  DZ(8),  FLUXA (576,8),  DOSEA (576,8),         0006. R55
      1           FLUXE (320,8),  S(8,8), SRZ(8),  A(8),                 0007 R55
      2           SS(576,8)    , FD(576), SOS (18,4,8),                  0008 R55
      3           FAD (576,8)  , FLIB(5), RZ(8)                          0009 R55
        COMMON   Z ,  DZ ,  FLUXA ,  DOSEA ,  FLUXE ,  S , SRZ ,         0010 R55
      1          A ,  SS ,  FD ,  SOS ,  FAD , FLIB ,  RZ                0011 R55
   10 CALL LIB1 (M)                                                      0012 R55
      GO TO (11,50) , M                                                  0013 R55
   11 CALL LIB (3HR55,ED)                                                0014 R55
      READ 500, FLIP                                                     00151R55
       IF ( FLIP ) 14, 14, 13                                           00152R55
   13 BACKSPACE 9                                                        00153R55
      GO TO 50                                                           0016 R55
   14 FLIP  =   MODF ( ED , 10.0)                                        0017 R55
      LIBT  =   FLIP                                                     0018 R55
      GO TO (15,20,25,30,35) , LIBT                                     0019. R55
   40 FORMAT (6E10.4)                                                    0020 R55
   15 READ 40,  (  Z(J), J=1,8 )                                         0021 R55
      XID1 = ED                                                          0022 R55
      GO TO 11                                                           0023 R55
   20 READ 40,    (DZ(J), J=1,8 )                                        0024 R55
      XID2 =  ED                                                         0025 R55
      GO TO 11                                                           0026 R55
   25 READ 40,   ((FLUXA (I,J), J=1,8) I=1,576 )                         0027 R55
      XID3 =  ED                                                         0028 R55
      GO TO 11                                                           0029 R55
   30 READ 40,   ((DOSEA (I,J), J=1,8) I=1,576 )                         0030 R55
      XID4 =  ED                                                         0031. R55
      GO TO 11                                                           0032 R55
   35 READ 40,   ((FLUXE (I,J), J=1,8) I=1,320 )                         0033 R55
      XID5 =  ED                                                         0034. R55
      GO TO 11                                                           0035 R55
   50 CALL SETUP (3HR55,XID)                                             0036 R55
      READ 500,   (FLIB(N), N=1,5)                                       0037. R55
  500 FORMAT (5F7.0)                                                     0038 R55
      IF (XID1 - FLIB(1))    55, 51, 55                                  0039 R55
   51 IF (XID2 - FLIB(2))    55, 52, 55                                  0040 R55
   52 IF (XID3 - FLIB(3))    55, 53, 55                                  0041 R55
   53 IF (XID4 - FLIB(4))    55, 54, 55                                  0042 R55
   54 IF (XID5 - FLIB(5))    55, 58, 55                                  0043. R55
   55 PRINT 56                                                           0044 R55
   56 FORMAT (54H LIBRARY DECKS CALLED FOR IN PROBLEM ARE NOT AVAILABLE)0045 R55
      CALL END9                                                          0046. R55
   58 READ 59, (NOP1  , NOP2 , NOP3 )                                    0047 R55
   59 FORMAT (3I3)                                                       0048 R55
      IF (NOP1 + NOP2 + NOP3 )  1000, 1000, 60                           0049. R55
   60 READ 40,   ((S(I,J), J=1,8),I=1,8 )                                0050 R55
      DO 70  J=1,8                                                       0051 R55
      RZ(J) =  0.01745329   * Z(J)                                       0052 R55
      SRZ(J)=  SINF (RZ(J))                                              0053 R55
      A(J)  =  SRZ(J)   *  DZ(J)                                         0054 R55
   70 CONTINUE                                                           0055 R55
```

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 900 IF ( NOP1 + NOP2 ) 1000, 1100, 1200                        0056 R55
1000 PRINT 1001                                                 0057 R55
1001 FORMAT (37H ERROR ON SECOND CARD OF PROBLEM DECK)          0058 R55
     CALL END9                                                  0059 R55
1100 NM = 320                                                   0060 R55
     GO TO 2000                                                 0061 R55
1200 NM = 576                                                   0062 R55
C                                                               0063 R55
C       CONSTRUCT A LARGER MATRIX FROM S(I,J)                   0064 R55
2000 DO 2022   I=1,NM                                           0065 R55
     DO 2022   J=1,8                                            0066 R55
     IF (I-8)  2010, 2010, 2012                                 0067 R55
2010 SS(I,J)  =  S(I,J)                                         0068 R55
     GO TO 2022                                                 0069 R55
2012 SS(I,J)  =  SS (I-8, J)                                    00701R55
2022 CONTINUE                                                   00702R55
  65 FORMAT (8E10.4)                                            00703R55
     IF ( SENSE SWITCH   2 )   2023, 2024                       00711R55
2023 PRINT 65, (( SS(I,J), J=1,8 ),I=1,NM )                     00712R55
2024 DO    2027    I=1,NM                                       00713R55
     DO    2027    J=1,8                                        00714R55
     SS(I,J)  =  6.2831853  *  SS(I,J)                          00721R55
2027 CONTINUE                                                   00722R55
     IF (SENSE SWITCH 2 )   2028, 2029                          00723R55
2028    PRINT 65, ((SS(I,J), J=1,8 ), I=1, NM )                 00724R55
2029 IF  (NOP1 )  1000, 2150, 2030                              0073 R55
C                                                               0074 R55
C       COMPUTE ANGULAR DISTRIBUTION OF FLUX.                   0075 R55
2030 DO 2035   I= 1,576                                         0076 R55
     DO 2035   J= 1,8                                           0077 R55
     FAD (I,J)  =   SS(I,J)  *  FLUXA (I,J)                      0078 R55
     FAD (I,J)  =  FAD(I,J)  *   A(J)                           0079 R55
2035 CONTINUE                                                   0080 R55
C    SUMMATION                                                  0081 R55
     DO 2040  I=1,576                                           0082 R55
     FD(I) = 0.0                                                0083 R55
     DO 2040  J=1,8                                             0084 R55
     FD(I) = FD(I)   +   FAD (I,J)                              0085 R55
2040 CONTINUE                                                   0086 R55
     DO 2050  I=1,18                                            0087 R55
     DO 2050  J=1,4                                             0088 R55
     DO  2050  K=1,8                                            0089 R55
     L = K  +  8* (J-1)  +  32 * (I-1)                          0090 R55
     SOS (I,J,K)  =  FD(L)                                      0091 R55
2050 CONTINUE                                                   0092 R55
C    PRINT OUT RESULTS AS 4  18X8 MATRICES.                     0093 R55
     DO 2100  J=1,4                                             0094 R55
     PRINT 2060                                                 0095 R55
2060 FORMAT (29H1ANGULAR DISTRIBUTION OF FLUX)                  0096 R55
     PRINT 2070 , J                                             0097 R55
2070 FORMAT (22HOSEPARATION DISTANCE A,I1)                      0098 R55
     PRINT 2080                                                 0099 R55
2080 FORMAT (100HO K     E01       E02       E03     E04        0100 R55
    1 E05       E06       E07       E08     )                   0101 R55
     DO 2100 I=1,18                                             0102 R55
     PRINT 2090 , (I, (SOS (I,J,K), K=1,8 ))                    0103 R55
2090 FORMAT  (1H ,I2,8E12.4)                                    0104 R55
2100 CONTINUE                                                   0105 R55
```

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 2150 IF (NOP 2 )  1000, 3000, 2200                                    0106 R55
C                                                                      0107 R55
C       COMPUTE ANGULAR DISTRIBUTION OF DOSE RATE.                     0108 R55
 2200 DO 2210   I = 1,576                                              0109 R55
      DO 2210   J = 1,8                                                0110 R55
      FAD (I,J) = SS(I,J)  *  DOSEA  (I,J)                             0111 R55
      FAD (I,J) = FAD(I,J) *  A(J)                                     0112 R55
 2210 CONTINUE                                                         0113 R55
C     SUMMATION                                                        0114 R55
      DO 2220   I=1,576                                                0115 R55
      FD(I)  = 0.0                                                     0116 R55
      DO 2220   J=1,8                                                  0117 R55
      FD(I) = FD(I)  +  FAD (I,J)                                      0118 R55
 2220 CONTINUE                                                         0119 R55
      DO 2230   I=1,18                                                 0120 R55
      DO 2230   J=1,4                                                  0121 R55
      DO 2230   K=1,8                                                  0122 R55
      L  =  K  +  8 * (J-1)  +  32 * (I-1)                             0123 R55
      SOS(I,J,K)  =  FD(L)                                             0124 R55
 2230 CONTINUE                                                         0125 R55
C     PRINT OUT ANGULAR DISTRIBUTION OF DOSE RATE                      0126 R55
      DO 2250  J=1,4                                                   0127 R55
      PRINT 2240                                                       0128 R55
 2240 FORMAT (38H1ANGULAR DISTRIBUTION OF THE DOSE RATE)               0129 R55
      PRINT 2070,  J                                                   0130 R55
      PRINT 2080                                                       0131 R55
      DO 2250   I=1,18                                                 0132 R55
      PRINT 2090 ,  (I , (SOS (I,J,K), K=1,8) )                        0133 R55
 2250 CONTINUE                                                         0134 R55
      IF (NOP 3 )  1000,  50, 3000                                     0135 R55
C                                                                      0136 R55
C     COMPUTE ENERGY SPECTRUM OF FLUX                                  0137 R55
 3000 DO 3035   I=1,320                                                0138 R55
      DO 3035   J=1,8                                                  0139 R55
      SS(I,J) = SS(I,J)  * FLUXE (I,J)                                 0140 R55
      SS(I,J) = SS(I,J)  * A(J)                                        0141 R55
 3035 CONTINUE                                                         0142 R55
C     SUMMATION                                                        0143 R55
      DO 3040   I=1,320                                                0144 R55
      FD(I)  = 0.0                                                     0145 R55
      DO 3040   J=1,8                                                  0146 R55
      FD(I)  = FD(I) +  SS (I,J)                                       0147 R55
 3040 CONTINUE                                                         0148 R55
      DO 3050   I= 1,10                                                0149 R55
      DO 3050   J= 1,4                                                 0150 R55
      DO 3050   K= 1,8                                                 0151 R55
      L  =  K  +  8 * (J-1)  +  32 * (I-1)                             0152 R55
      SOS (I,J,K )  = FD(L)                                            0153 R55
 3050 CONTINUE                                                         0154 R55
C     PRINT OUT ENERGY SPECTRUM OF FLUX                                0155 R55
      DO 3070  J=1,4                                                   0156 R55
      PRINT 3060                                                       0157 R55
 3060 FORMAT (29H1TOTAL FLUX IN ENERGY GROUP K)                        0158 R55
C                                                                      0159 R55
      PRINT 2070 , J                                                   0160 R55
      PRINT 2080                                                       0161 R55
      DO 3070  I=1,10                                                  0162 R55
      PRINT 2090 , ( I,  (SOS(I,J,K),K=1,8) )                          0163 R55
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3070 CONTINUE                                    0164 R55
4000 GO TO 50                                    0165 R55
     END(2,1)                                    0166 R55
```

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